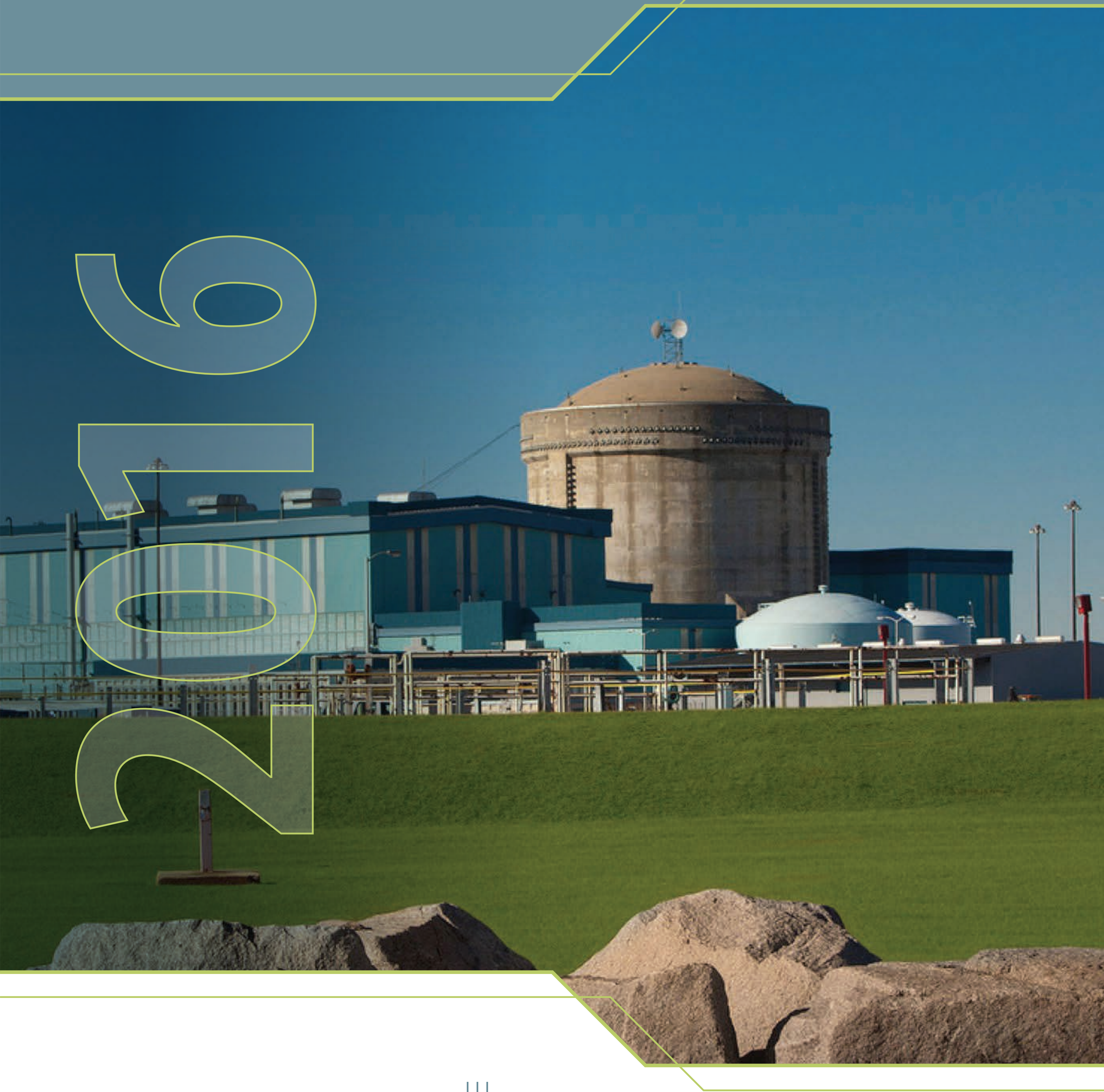
Light Water Reactor Sustainability Program

ACCOMPLISHMENTS REPORT



2016

LWRS
Light Water Reactor Sustainability

U.S. DEPARTMENT OF ENERGY

From the LWRS Program Technical Integration Office Director

Kathryn A. McCarthy, Director, LWRS Program Technical Integration Office

Welcome to the 2016 Light Water Reactor Sustainability (LWRS) Program Accomplishments Report, covering research and development (R&D) highlights from 2016. The LWRS Program is a U.S. Department of Energy (DOE) R&D program to inform and support the long-term operation of our nation's commercial nuclear power plants. The research uses the unique facilities and capabilities at the DOE national laboratories in collaboration with industry, academia, and international partners.

Extending the operation of current nuclear power plants is essential to supporting our nation's base load energy infrastructure with clean and affordable energy. The purpose of the LWRS Program is to provide technical results for plant owners to make informed decisions on maintaining their plants through periods of extended (long-term) operation and seeking renewed operating licenses for the plants from the U.S. Nuclear Regulatory Commission (NRC), reducing the uncertainties and accompanying risks associated with those decisions.

In January 2013, 104 nuclear power plants operated in 31 states. However, since then, five plants have been shut down (most due to economic reasons) with additional shutdowns planned or under consideration. The LWRS Program is aimed at sustaining the existing fleet of commercial reactors with R&D products (models and technologies) that, when implemented, provide tangible economic benefits as well as data that can be used by industry and NRC for license renewal decisions. The LWRS Program continues to work closely with the Electric Power Research Institute (EPRI) to ensure that our efforts address high priority industry needs.

As the LWRS Program finishes its eighth year, we're proud that many of our R&D products are already being used by industry. This includes technologies that decrease plant operation and management costs as well as data that is informing decisions on license renewal. The LWRS Program's partnerships with plant owners/operators grow each year and are essential to our success.

This report covers selected highlights from the four research pathways in the LWRS Program: Materials Aging and Degradation; Risk-Informed Safety Margin Characterization; Advanced Instrumentation, Information, and Control Systems Technologies; and Reactor Safety Technologies, as well as a look-ahead at planned activities for 2017. If you have any questions about the information in the report or about the LWRS Program, please contact me, Richard A. Reister (the DOE Federal Program Manager), or the respective research pathway leader (noted on pages 46 and 47), or visit the LWRS Program website (https://lwrs.inl.gov). The annually updated Integrated Program Plan and Pathway Technical Program Plans are also available on the LWRS website for those seeking more technical information on LWRS Program R&D projects.

The Brunswick Nuclear Generating Station is a 2-unit nuclear power plant near Southport, North Carolina, is a partner in an LWRS Program pilot project on the Advanced Outage Control Center.

The mission of the Light Water Reactor Sustainability Program is development of the scientific basis, and science-based methodologies and tools, for the safe and economical long-term operation of the nation's high-performing fleet of commercial nuclear energy facilities.

Contents

On the Cover

The Virgil C. Summer Nuclear Generating Station near Jenkinsville, South Carolina, is a partner in an LWRS Program pilot project on developing a digital architecture for information technology.

INTRODUCTION

Nuclear power has safely, reliably and economically contributed almost 20% of the total electrical generation in the United States over the past two decades, and it remains the single largest contributor (more than 60%) of U.S. non-greenhouse-gas-emitting electric power generation. Operation of the existing fleet of commercial nuclear power plants to 60 years, extending the operating lifetimes of those plants beyond 60 years and, where practical, making further improvements in their productivity are essential to the plants continuing as a key domestic source of dependable and affordable base load electrical energy.

The Light Water Reactor Sustainability (LWRS) Program is a research and development (R&D) program sponsored by the U. S. Department of Energy (DOE) and performed in cooperation with the related R&D programs of the nuclear industry and the U.S. Nuclear Regulatory Commission (NRC). The LWRS Program provides technical foundations for licensing and managing the long-term safe and economical operation of current nuclear power plants, utilizing the unique capabilities of the DOE national laboratory system.

The LWRS Program has two facets with respect to long-term operations: (1) understand and manage the aging of nuclear power plant systems, structures, and components (SSCs) and how to best manage them so that the plants can continue to operate safely, efficiently and economically; and (2) provide science-based solutions to the industry for exceeding the performance of the current labor-intensive business model. The program's R&D role focuses on aging phenomena and issues that require long-term research and/or unique DOE laboratory expertise and facilities and are applicable to a broad range of operating reactors. When appropriate, R&D activities are cost shared with industry and/or NRC. Pilot projects and collaborative activities are underway at commercial nuclear facilities and with industry organizations. The LWRS Program is coordinated with the Electric Power Research Institute and its Long-Term Operations Program.

In addition to long-term operations broadly, the LWRS Program also supports second license renewal activities. Two owner/operators have announced their intent to apply to the NRC for a second 20-year license renewal (i.e., operation of the plants for up to 80 years). The LWRS Program will work with these organizations, as well as other owner/ operators who intend to apply for a second license renewal, to provide the technical basis for second license renewal specifically, and long-term operations generally.

The LWRS Program consists of the following four primary R&D technical areas:

The LWRS Program provides technical foundations for managing the long-term safe and economical operation of current nuclear power plants, utilizing the unique capabilities of the DOE national laboratory system.

Materials Aging and Degradation

Nuclear reactors present a very challenging service environment. Extending reactor service lifetimes up to and beyond 60 years increases the operational demands on materials and components. Materials research provides an important foundation for licensing and managing the long-term safe and economical operation of nuclear power plants. The strategic goals of the Materials Aging and Degradation Pathway are to develop the scientific basis for understanding and predicting long-term environmental degradation behavior of materials in nuclear power plants and to provide data and methods to assess performance of SSCs essential to safe and sustained nuclear power plant operations.

This welding cubicle was installed in a hot cell for irradiated materials welding technique development and testing.

Key research results to date in this technical area include:

- Completed the Expanded Materials Degradation Assessment (project co-funded with NRC), informing an update to NRC's Generic Aging Lessons Learned Report that will aid NRC in its deliberations on applications by licensees for a second renewal of reactor operating licenses.
- Expanded the knowledge base on aging of nuclear facility concrete structures through the release of a nuclear concrete database with never-before-published data on concrete behavior under irradiation – data that will be used in license renewal applications to justify conclusions made by licensees on the adequacy of plant concrete structures for long-term operations.
- Analyzed the remaining useful life of service-aged cabling (Anaconda Densheath EPR Cable; 40 plus years of service in the Oak Ridge National Laboratory's High Flux Isotope Reactor) that is representative of cables in commercial plants – results showed remaining useful life well in excess of 80 years; this information will also help to inform additional R&D studies in progress on methods to better predict remaining useful life of cables installed in the current U.S. reactor fleet.
- In collaboration with the Electric Power Research Institute and Areva, analyzed a failed-in-service component (Alloy 718 hold down springs in fuel assemblies) and identified failure mechanisms for consideration in implementing needed corrective measures.
- Completed the design and construction of a welding cubicle for developing advanced welding techniques for irradiated material.

High-level planned accomplishments in the near term include:

- Providing mechanistic understanding of key materials degradation processes, predictive capabilities, and high-quality data to inform decisions and processes by both industry and regulators, including:
  - Predictive models for swelling in light water reactor (LWR) components, aging of cast austenitic stainless steel components, cable degradation, and nickel-base alloy stress corrosion cracking susceptibility;
  - Models for transition temperature shifts in reactor pressure vessel steels, precipitate phase stability and formation in Alloy 316, and environmentally assisted fatigue in LWR components;

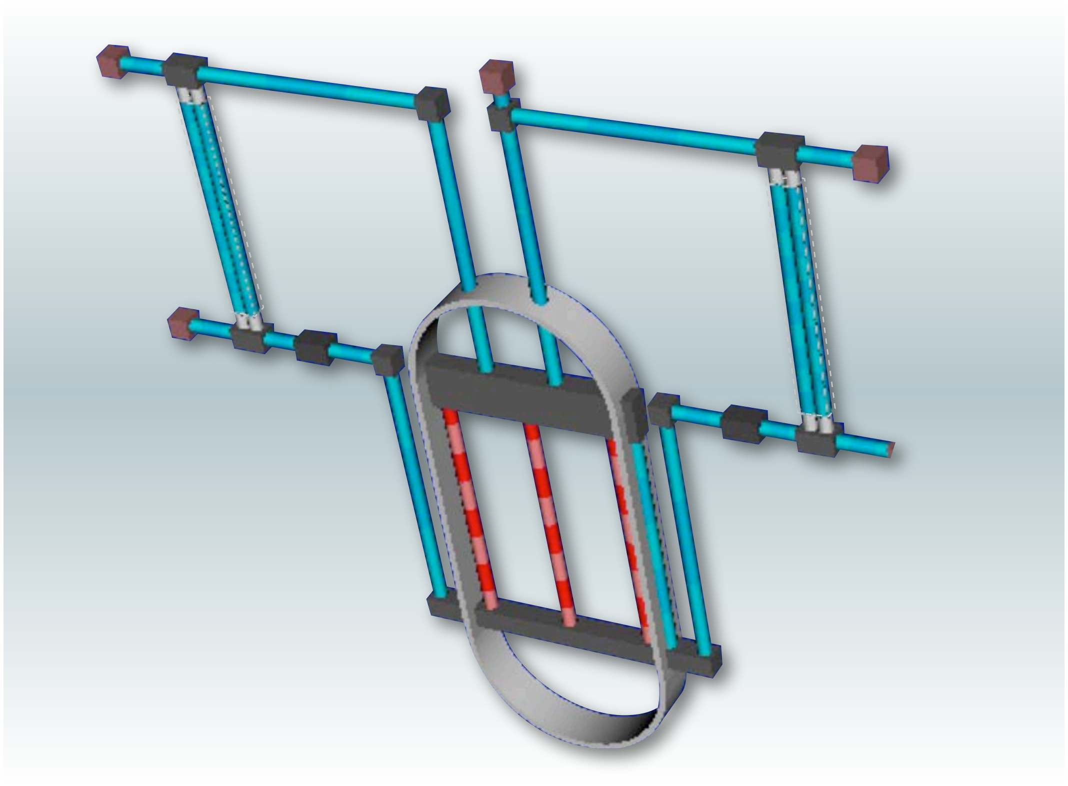
"Taking part in the pilot group and testing the first implementation of an electronic work order was a great opportunity to help site maintenance step into the future. The system is very user friendly and has endless applications. With the feedback provided by myself and my co-workers, electronic work orders will increase the efficiency of turnovers and status updates by providing real time information directly from the field, as well as ease the administrative burden of the frontline."

- Palo Verde Nuclear Generating Station Mechanical Team Member

- Prototype proof-of-concept system for nondestructive examination of concrete sections, fatigue damage, and cable insulation;
- Harvesting of reactor pressure vessel materials, cable and baffle former bolt components for examination of in-service materials for model development and comparison to high flux data; and
- Development and transfer of weld repair technique for welding irradiated materials to industry.

Risk-Informed Safety Margin Characterization (RISMC)

Safety is central to the design, licensing, operation, and economics of nuclear power plants. As the current LWR fleet continues operation up to and beyond 60 years, there are possibilities for increased frequency of SSC failures that initiate safety-significant events, reduce existing accident mitigation capabilities, impact plant operation, or create new failure modes. The RISMC Pathway provides an enhanced understanding of LWR safety by developing methods, tools, and data in support of risk-informed margins management. The purpose of the RISMC Pathway R&D is to support plant decisions for risk-informed margins management with the aim to improve the economics and reliability and sustain the safety of current nuclear power plants over periods of extended plant operations. The goals of the RISMC Pathway are twofold: (1) develop and demonstrate a risk-assessment method that is coupled to safety margin quantification that can be used by nuclear power plant decision makers as part of risk-informed margin management strategies; (2) create an advanced RISMC Toolkit that



RELAP-7 is the main reactor systems simulation tool for RISMC and the next-generation in the RELAP reactor safety/ systems analysis application series.

enables more accurate representation of nuclear power plant safety margins and their associated impact on operations and economics.

Key research results to date include:

- Demonstrated the RISMC methodology, using the newly developed RELAP-7 systems analysis code, by application to a nuclear power plant station blackout scenario – uses of the methodology will increase to address additional industry performance topics using an expanded set of safety analysis tools.
- Demonstrated the application of the Grizzly component aging simulation code plus the RAVEN probabilistic analysis tool to probabilistic fracture mechanics analysis of crack initiation in a reactor pressure vessel under a pressurized thermal shock transient – the Grizzly code is an engineering tool that when completed, can be applied to study a variety of degradation mechanisms in nuclear power plant components.
- Demonstrated external hazards (flooding and seismic) analysis methods and tools including door flooding fragility experiments – these tools will provide more accurate analysis of flooding and seismic scenarios, providing an opportunity to reduce the conservatism present in today's analyses.

High-level planned accomplishments in the near term include:

- Margins analysis techniques and associated models and tools to enable industry to conduct margins quantification exercises for their plants, including:
  - Demonstration of the margins analysis techniques on industry-important topics (performance-based emergency core cooling system cladding acceptance criteria, external hazard analyses, reactor containment analysis, and long-term coping studies);
  - A modern, validated safety analysis tool (RELAP-7);
  - Modern, validated flooding and seismic analysis tools;
  - Component aging and damage evolution analysis tool (Grizzly), capable of modeling aging of select steel (embrittlement) and concrete failure mechanisms; and
  - An advanced probabilistic and data mining analysis tool (RAVEN).

Advanced Instrumentation, Information and Control (II&C) Systems Technologies

Reliable instrumentation, information, and control (II&C) systems technologies are essential to ensuring safe and efficient operation of the U.S. commercial reactor fleet. Replacing existing analog systems with digital technologies has not been undertaken to a large extent within the nuclear power industry worldwide due to significant technical and regulatory uncertainty. The Advanced II&C Systems Technologies Pathway conducts targeted R&D to address aging and reliability concerns with the legacy instrumentation and control and related information systems of the U.S. LWR fleet. This work involves two major goals: (1) to ensure that legacy analog II&C systems are not life-limiting for the LWR fleet, and (2) to implement digital II&C technology in a manner that enables broad innovation and business improvement in the nuclear power plant operating model. Technologies are developed and tested via pilot projects at nuclear power plants, together with plant personnel.

Key research results to date include:

- In collaboration with industry, developed and demonstrated advanced outage control center technologies that received a Nuclear Energy Institute Top Industry Practice award in 2014; the utility receiving the Top Industry Practice award cited a $48 million cost savings due to reduced outage time. This technology is now being deployed in several nuclear power plants.
- In collaboration with industry, developed and demonstrated computer-based procedures that are being implemented at several nuclear power plants.
- Developed and demonstrated a methodology to analyze the business case for digital upgrades;
  - An analysis of mobile work packages showed approximately $6.5M in annual savings, representing a net present value of over $21M through the expected 15-year life of the technology.
  - An analysis of advanced outage management showed an annual savings of greater than $7.7M/year, with a present value over $48M through the expected 15-year life of the technology.
- Completed the assembly of the Human Systems Simulation Laboratory, a user facility at Idaho National Laboratory (INL), and demonstrated its capability to model a nuclear power plant control room. The Human Systems Simulation

Reactor operators test a new prototype digital interface in the Human Systems Simulation Laboratory.

Laboratory is being used to evaluate cost effective plant modernization strategies that can keep existing plants economically competitive.

High-level planned accomplishments in the near term include the production of guides to implement digital technologies, including:

- Hybrid integrated control room incorporating digital upgrades in an analog control room, advanced alarm systems, and control room computer-based procedures;
- Digital architecture for an automated plant;
- Human performance improvement for nuclear power plant field workers including mobile technologies for nuclear power plant field workers, and automated work packages;
- Advanced online monitoring facility for integrated operations;
- Outage safety and efficiency including advanced outage coordination, advanced outage control center, and outage risk management improvement; and
- Online monitoring of passive components.

Reactor Safety Technologies

In the aftermath of the March 2011 multi-unit accident at the Fukushima Daiichi nuclear power plant in Japan, the nuclear community has been reassessing certain safety assumptions about nuclear reactor plant design, operations and emergency actions, particularly with respect to extreme events that might occur and that are beyond each plant's current design basis. The Reactor Safety Technologies Pathway goals are to improve understanding of beyond design basis events and reduce uncertainty in severe accident progression, phenomenology, and outcomes using existing analytical codes and information gleaned from severe accidents, in particular the Fukushima Daiichi events. This information will be used to aid in developing

An overview display will be developed to allow advanced outage control center management to quickly evaluate outage status.

Region	Examination Information Classification			
	Visual	Near-Proximity	Destructive	Analytical
Reactor Building				
Reactor Core Isolation Cooling	****	***	**	
High Pressure Core Injection	****		***	
Building	****	***	**	*
Primary Containment Vessel				
Main Steam Line and Safety Relief Valves	****		***	
Drywell Area	****	***	**	*
Suppression Chamber	****	***		
Pedestal / Reactor Pressure Vessel - Lower Head	****		***	**
Instrumentation		****	***	
Reactor Pressure Vessel				
Upper Vessel Penetrations	****		***	**
Upper Internals	****	***	**	*
Core Regions and Shroud	****		***	**
Lower Plenum	****		***	**

Fukushima Daiichi forensic examinations recommendations (highest priority activities are those with the most asterisks).

mitigating strategies and improving severe accident management guidelines for the current LWR fleet.

Key research results to date include:

- Completed a report documenting recommendations for forensics examinations in the Fukushima Daiichi nuclear power plants; information from these forensics examinations will improve the understanding of severe accident progression.
- Completed a comparison of the MELCOR and MAAP severe accident codes modeling of the Fukushima Daiichi events, providing information that could be used to improve these codes.

High-level planned accomplishments in the near term include:

- Improved understanding of and reduced uncertainty in severe accident progression, phenomenology, and outcomes, including:
  - Forensics inspection plan for Fukushima-Daiichi reactors; and
  - Reactor core isolation cooling system performance.

The key accomplishments of the LWRS Program in 2016 are summarized in the following pages of this report. A more complete summary of program R&D efforts, including accomplishments and near-term performance milestones can be found in the LWRS Program Integrated Program Plan posted on the LWRS Program Web site, https://lwrs.inl.gov.

2016 RESEARCH HIGHLIGHTS

Materials Aging and Degradation

Research and development efforts in this pathway are developing the scientific basis for understanding and predicting long-term behavior of materials in nuclear power plants. This work will inform long-term operation decisions generally, and second license renewal decisions specifically, by providing data and methods to assess the performance of systems, structures, and components essential to safe and sustained nuclear power plant operations. This includes methods for monitoring and assessing degradation via nondestructive techniques, and strategies for mitigating the effects of aging.

Research Highlights

The research and development in this pathway falls into five categories: reactor metals, concrete, cables, mitigation technologies, and cross-cutting research activities such as harvesting materials from operating and decommissioned power plants. Select research and development highlights are provided here, followed by a list of major accomplishments (detailed reports covering the accomplishments can be found on the LWRS Program website, (https://lwrs.inl.gov).

Reactor Metals

Numerous metal alloys can be found throughout the primary and secondary reactor systems. Some of these materials (in particular, the reactor core internals) are exposed to high temperatures, water, and neutron fluence. This challenging operating environment creates degradation mechanisms in the materials that are unique to nuclear reactor service.

Reactor Metals Highlight:

Mechanisms of Irradiation Assisted Stress Corrosion Cracking. Austenitic AISI 304 and 316 stainless steels, as well as their numerous variants, are widely employed in the nuclear industry. These steels were chosen in the early 1960s because of their favorable combination of mechanical and corrosion properties, machinability, weldability, price, and—at that time—acceptable radiation tolerance. Recently, a number of different materials have come into use, but 300-series steels are expected to remain in service for at least the next 20 to 30 years, if not longer.

Although 300-series steels have an advantageous combination of properties, they are known to experience several service-related issues, one of which is irradiation-assisted stress corrosion cracking (IASCC). IASCC is one of the widely recognized and most severe concerns associated with this class of materials in light water reactor (LWR) operating environments. Recently, significant progress was made in mitigating IASCC by transitioning to hydrogen water chemistry, employing corrosion inhibitors, and decreasing corrosion potential, among other methods. However, IASCC-related issues are still expected to become more severe as nuclear power plants and their components operate in extended service lifetime environments.

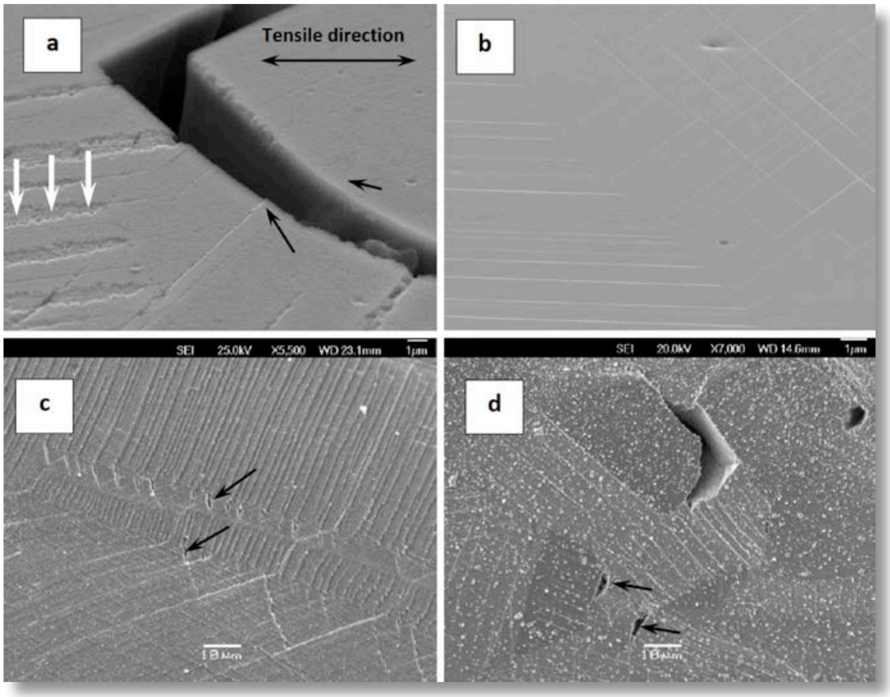
In this work done at Oak Ridge National Laboratory (ORNL), a methodical multiscale and multi-tool approach was used to characterize the corrosion processes and localized deformation in neutron-irradiated materials subjected to testing in a corrosion environment. This work takes advantage of coupling several analytical techniques, including SEM-EBSD (scanning electron microscopy-electron backscatter diffraction), FIB (focused ion beam), site-specific TEM (transmission electron microscopy) sample preparation, and STEM (scanning transmission electron microscopy). Key to this work was the use of novel and/or state-of-the-art techniques and equipment to understand effects that have been difficult or impossible to uncover to date. This includes the use of a high-resolution, high-efficiency STEM-based EDS (energy dispersive x-ray spectroscopy) spectrum mapping equipment/technique coupled with targeted FIB lift-out procedures based on extensive SEM-EBSD–based investigations.

The results highlight the overall complexity of IASCC in irradiated stainless steels. It was found that the initiation of IASCC involves many factors including localized deformation and most likely selective oxidation at the surface steps produced by channels penetrating the free surface. Grain orientation was found to dictate the location of observable small cracks at the specimen surface. The role of grain orientation was determined to be more complex than previously thought. Finally, it was found that cracks could exhibit a complex structure and chemistry, with multilayer oxide formation and variations in oxide formation occurring with increasing depth in the crack. This work supports delivery of a predictive model for IASCC susceptibility in 2019.

Reactor Metals Highlight:

High fluence Reactor Pressure Vessel Steel. The reactor pressure vessel (RPV) is a critical component in commercial nuclear reactors; therefore, understanding the property

An enlarged view of a crack in A-alloy, which has formed along a grain boundary and slip lines at the sample surface (which indicate localized deformation) are shown in (a). Black arrows point to surface steps caused by dislocation channels. White arrows show surface erosion along the channels. Local strain is ~1.3%. (b) Surface relief example for non-irradiated commercial 304 steel strained at 1% at RT and strain rate 10–3s–1 (for comparison). Images (c) and (d) are examples of transgranular cracks (pointed by arrows) for (c) SW- and (d) A-alloys. SW- and A-alloys are AISI 304 stainless steels.



changes as a result of radiation exposure and radiation-induced microstructural changes is important. The so-called "late-blooming phases" of Mn-Ni-Si enriched particles, especially for high-nickel welds, have been observed, and additional experimental data are needed in the high fluence regime where they are expected. The ATR-2 irradiation experiment involves examining materials property changes in RPV alloys of varying composition and includes archival materials used in commercial plant surveillance materials. The target fluence for the ATR-2 experiment ranged from the equivalent of 40 to 80 years of LWR operation and bridges a flux-fluence gap in available data. Atom probe tomography (APT) was performed on a series of surveillance steels with varying bulk Cu, Ni, Mn, and Si contents irradiated to a fluence of approximately 1.4×10^{20} n/cm2 at approximately 290°C by the University of California-Santa Barbara as part of the ATR-2 experiment.

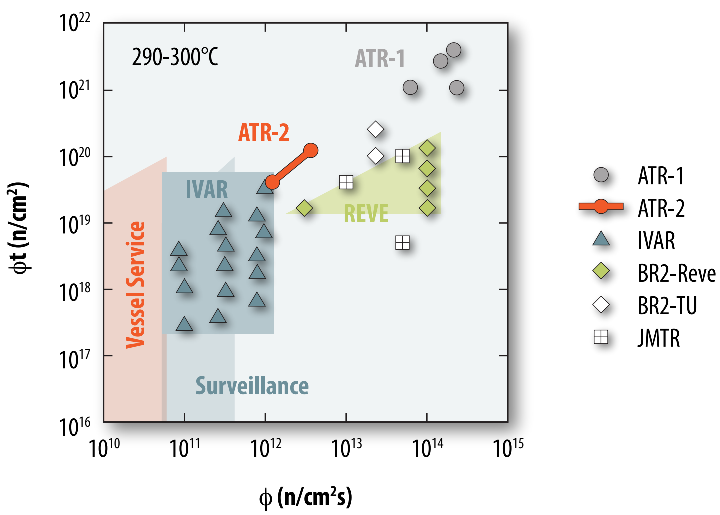
APT was also carried out at ORNL. Analysis efforts included quantifying precipitate volume fractions, sizes, number densities and compositions that formed under irradiation. Work included the mapping of precipitate size distributions, quantifying segregation of Cu/Ni/Mn/Si/P to microstructural features and examining precipitate nucleation on dislocations and loops. This work supports completion of a validated model for transition temperature shifts in RPV steels in 2018.

Concrete

Many concrete-based structures are part of a typical LWR plant, such as the foundation, support, shielding, and containment. Concrete has been used in nuclear power plant construction because of its structural strength, ability to shield radiation, ease of fabrication, and low cost. Examples of concrete structures important to LWR safety include the containment building, the spent fuel pool, and cooling towers. As concrete ages, changes in properties occur as a result of continuing microstructural changes (e.g., slow hydration, crystallization of amorphous constituents, and reactions between cement paste and aggregates), as well as environmental influences. Further changes are predicted due to interactions with radiation fields.

Concrete Highlight:

Alkali-Silica Reaction Mockup. A mode of degradation being evaluated for its impact on



The ATR-2 irradiation experiment involves examining materials property changes in RPV alloys with a target fluence the equivalent of 40 to 80 years of LWR operation.

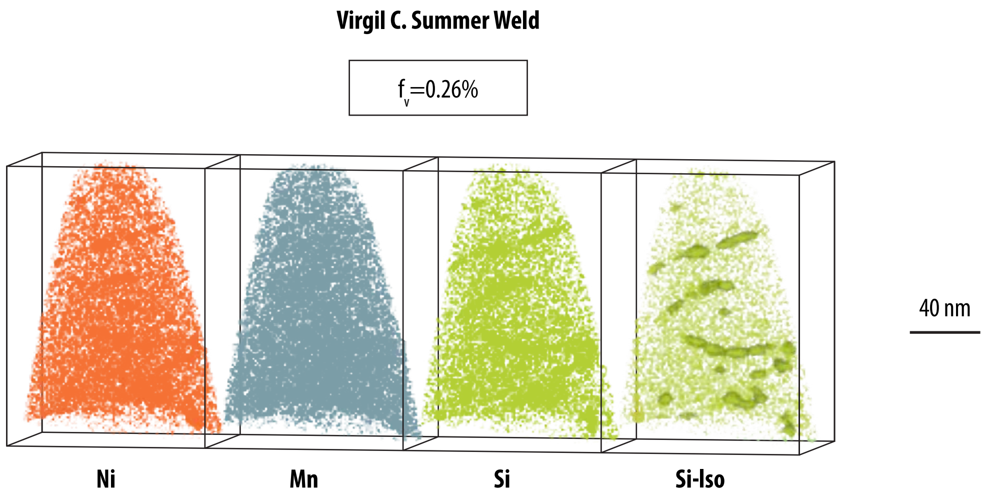
structural concrete performance is that of alkali-silica reactions (ASR) that can produce swelling of the concrete paste, resulting in cracking and weakening of the shear capacity of the concrete structure. This activity, underway at the University of Tennessee in conjunction with ORNL, studies the development of ASR expansion and induced damage of large-scale specimens representative of structural concrete elements found in nuclear power plants through experimentally validated models that explore the structural capacity of ASR affected structures like the biological shield, the containment building and fuel handling building.

Three specimens, corresponding to 136 x 116 x 40 in. reinforced concrete blocks, were cast and enclosed in an environmental chamber under specific temperature and relative humidity conditions to accelerate the ASR-induced expansion. Specimens 1 and 2 have been fabricated using highly reactive aggregates from North Carolina, according to a mix design study conducted by the University of Alabama. Specimen 1 was confined in a relatively rigid steel frame. Specimen 3, considered as a control specimen, was made with the same reactive aggregates, but the ASR has been mitigated by incorporating lithium in the mix.

Multiple sensors are embedded in the specimens to enable measurement of temperature, concrete strain, rebar strain, air temperature and humidity inside and outside the chamber, deformation in the Z-direction, expansion deformation, and acoustic data.

Although the residual shear capacity of the ASR-affected stress-confined concrete is one of the research objectives, the deterioration of several mechanical properties of the concrete as a function of ASR expansion is also being investigated as a means to develop and improve reliable condition assessment methodologies including ASR models. The characterization of mechanical properties as a function of the stress-confined ASR expansion will be used to validate and improve models to more accurately predict the effects of confinement on the development of ASR damage in nuclear structures.

Experimental testing will be conducted in accelerated conditions, employing extensive monitoring and nondestructive techniques to evaluate structural stresses generated in the large block test specimens. An example of the testing includes the ASR Test Assembly (shown on page 16), which will provide an opportunity to monitor the development of ASR under accelerated conditions in very large representative structures. The development



Atom Probe Tomography (APT) elemental maps of Mn-Ni-Si precipitates along several dislocations in the low Cu/high Ni (0.04wt% Cu, 1.0wt% Ni) Virgil Summer Weld.

The recently completed ASR mockup specimens are being used to study the development of ASR expansion and induced damage of large-scale specimens representative of structural concrete elements found in nuclear power plants.

of ASR will be monitored by both passive and active nondestructive examination techniques. Following conclusion of the monitoring program, final destructive testing will be performed to address the question of the shear capacity of the ASR affected concrete. This activity supports the completion of modeling to assess the combined effects of radiation and ASR on structural performance for concrete components in 2020.

Cables

A variety of environmental stressors in nuclear reactors can influence the aging of low and medium electrical-power and instrumentation and control cables and their insulation. These environmental factors include temperature, radiation, moisture/humidity, vibration, chemical spray, mechanical stress, and oxygen present in the surrounding gaseous environment (usually air). Exposure to these environmental stressors can lead to degradation that, if not appropriately managed, could cause insulation failure, which could prevent associated components from performing their intended function.

Cable Highlight

Cable Aging and Degradation. Many changes have occurred in cable manufacturing and in the formulation of cable insulation and jacketing materials since the oldest currently operating commercial nuclear power plants in the United States were constructed in the late 1960's/early 1970's. Developing a predictive understanding of the aging and degradation of cable system materials installed in existing nuclear power plants is challenging since many of the materials are no longer manufactured and little information may be available about their formulations. Investigation of the aging behavior of these materials must rely on historical data, procurement of vintage materials from storage, and harvesting of cables either following plant closure or during cable replacement activities. Harvesting of installed cables provides information about the aging of materials in actual plant environments. The Electric Power Research Institute (EPRI) has been instrumental in providing new old stock cables and harvested cables in support of collaborative aging research. The cables provided represent the most common insulation and jacket materials found in nuclear power plants, as well as those from the most common cable manufacturers.

Pacific Northwest National Laboratory and ORNL are working collaboratively on cable aging activities. This year, they performed accelerated thermal aging at 80 and 100°C on cable jacketing material harvested from the Callaway and San Onofre plants. The San Onofre cable consisting of chlorosulfonated polyethylene (CSPE) jacket and cross-linked polyethylene (XLPE) insulation was never used in-service, but stored on site in a climate-controlled location since 2006. This material is representative of other cables used in the nuclear plant. The Callaway control rod cable is a Boston Insulated Wire (BIW) fabricated Hypalon Jacket (also, CSPE) with Ethylene propylene rubber (EPR) insulated wires that was used in-service for approximately 30 years.

Accelerated aging was performed in air at 80, 100 and 120°C, with periodic removal of samples for testing over a time period up to 100 days. Follow-on post-exposure performance testing was performed through indenter modulus and tensile testing. Comparisons were made between the Callaway and San Onofre cables to similar aged materials from the Sandia National Laboratories' Cable Repository of Aged Polymer Samples and EPRI 's Cable Polymer Aging Database. For the San Onofre cable, accelerated aging results were found similar to that of Gillen and coworkers accelerated aging data, suggesting little change in the polymer occurred following site storage for nearly 10 years. However, reduction in elongation at break with aging time is observed for the Callaway control rod cable that did see in-service exposures, but the remaining useful life estimated from these tests is compatible with license extension to 80 years. Further test data at additional temperatures (on-going work) is required for a full assessment of the degradation processes occurring (further refinement of the activation energy with a broader data set), which will also be performed through subsequent testing such as oxidation induction time and thermo-gravimetric analysis. This works supports the completion of a predictive model for cable degradation in 2019.

Advanced Weld Repair

Welding is extensively used in construction of nuclear reactor components and subsystems. The performance of weldments (including both weld metal and the adjacent heat affected zone) is critical to the safe and efficient operation of the nuclear reactor. Weldments frequently are the most susceptible locations for corrosion, stress-corrosion, and mechanical failures. Weld repairs are a potential method for mitigating cracking or degradation instead of component replacement. With extended lifetimes and increased repair frequency, these welds must be resistant to corrosion, irradiation, and other forms of

Cross sections of harvested Callaway BIW control rod cable (left) and San Onofre RB FWIII cable (right). These cables were tested to examine degradation under extended operation.

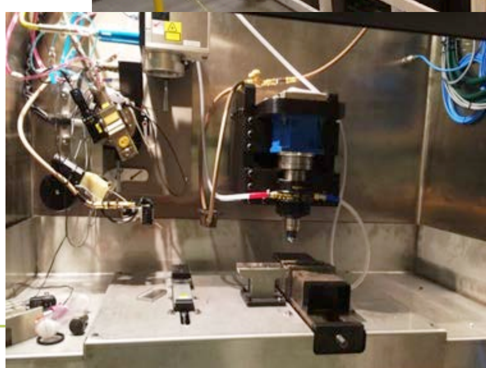
degradation. The LWRS Program is developing new welding and weld analysis techniques for welding highly irradiated material via a combination of experimental and modeling activities that are jointly funded with EPRI. Understanding the impact that helium, present in irradiated material, has on the welding process is an important input to development of an advanced technology.

Advanced Weld Repair Highlight:

Hot cell welding facility. To address the growing need for advanced weld repair technologies, specifically those that enable repair of highly irradiated materials, a hot cell welding facility was designed and installed at ORNL. This facility will enable researchers to identify, test, and validate the performance of the most promising technologies directly on irradiated specimens. The hot cell welding facility was developed through a partnership between the LWRS and EPRI LTO Programs. Following extensive weld process development and system design and construction, the primary components of the welding facility have now been successfully integrated into a hot cell at ORNL. The acceptable functionality of the integrated welding system was demonstrated recently with initial start-up testing of the friction stir welding system on unirradiated representative stainless steel materials. The system performed well, executing automated welding programs according to specifications. The successful installation of the integrated welding hot cell and demonstration of the baseline performance of the friction stir welding system establishes the foundation for a state-of-the-art DOE facility for research and development on welding repair of irradiated materials. This activity supports the transfer of a weld-repair technique to industry in 2018.

Harvesting Service Materials from Nuclear Reactors

Access to service materials from active or decommissioned nuclear reactors is invaluable because there is limited operational data or experience to inform decisions on extending nuclear reactor service lifetimes. In addition, access to service materials will facilitate coordination with other materials tasks, including an assessment of current degradation



The welding cubicle, enabling development and testing of advanced welding techniques on irradiated materials, is lowered into the hot cell. The friction stir weld system (inset photo) was successfully tested.

models to further develop the scientific basis for understanding and predicting long-term environmental degradation behavior.

Materials Harvesting Highlight

The decommissioning of the Zion Units 1 and 2 Nuclear Generating Station in Zion, Illinois, presents a special and timely opportunity for developing a better understanding of materials degradation and other issues associated with service lifetimes of existing nuclear power plants beyond 60 years. ORNL is coordinating and contracting with Zion Solutions, LLC, a subsidiary of Energy Solutions, for the selective procurement of materials, structures, and components, from the decommissioned reactors.

Under this activity, segments of the Zion Unit 1 RPV were obtained to evaluate potential degradation issues associated with extended operations. Four RPV segments were sent to the Energy Solutions Memphis Processing Facility, and cut into seven blocks: two from the beltline weld and five from base metal. The blocks will be shipped to BWXT for machining into test specimens for laboratory testing. The blocks will be machined into mechanical test specimens and microstructural characterization samples at BWXT in 2017. Access to service-irradiated RPV welds and plate sections will allow through wall attenuation studies to be performed, which will be used to assess current radiation damage models.

Segments from the Zion Unit 1 RPV were harvested to provide source material for laboratory testing (photo courtesy of Energy Solutions).

2016 Materials Aging and Degradation Accomplishments

A summary of the 2016 Materials Aging and Degradation Pathway accomplishments is provided below. For each research area, the major 2016 accomplishments follow the primary out-year deliverable that they support.

Reactor Metals

- Validated model for transition temperature shifts in reactor pressure vessel steels (2017)
  - Harvested two segments from the Zion Unit 1 RPV
  - Began post-irradiation examination of reactor pressure vessel alloys from the ATR-2 experiment
  - Developed alkali-silica model and incorporate into Grizzly
- Predictive model capability for nickel-base alloy stress corrosion cracking susceptibility (2019)
  - Identify the initiation mechanisms leading to stress corrosion crack development in Alloy 600 and 690 materials
- Model for environmentally assisted fatigue in LWR components (2017)
  - Completed cyclic plasticity material modeling of 508 low alloy steel and 316 stainless steel under stress control and variable loading environmental fatigue testing
- Predictive model capability for IASCC susceptibility (2019)
  - Developed computational tools to model thermodynamics and kinetics of thermal- and radiation-induced segregation in Fe-Cr-Ni austenitic steels
  - Retrieved high fluence baffle bolts from the R.E. Ginna nuclear power plant

- Predictive capability for cast stainless steel components under extended service conditions (2018)
  - Investigated the mechanical performance of model cast austenitic steels following 1,500 hours aging
  - Characterized materials property changes of wrought 304L and 316L steels following 1,500 hours aging and EPRI cast austenitic stainless steels after 10,000 hours of aging

Cables

- Predictive model for determining end of useful life for cable insulation (2019)
  - Characterized oxidation of cross-linked polyethylene and ethylene propylene rubber insulation materials
  - Evaluated thermal aging of control rod cable at temperatures below 100°C
  - Harvested cables from Zion Unit 2
  - Analyzed ethylene propylene rubber degradation through accelerated aging testing
  - Completed combined thermal/radiation aging of harvested cable jacket
  - Evaluated bulk electrical non-destructive examinations for cable aging management
  - Harvested Crystal River Unit 3 cables and developed test plans

Concrete

- Tool to assess the combined effects of irradiation and alkali-silica reactions on structural performance for concrete components (2020)
  - Examined the effect of structural restraints and creep on radiation induced volumetric expansion rates in concrete
  - Constructed alkali-silica reaction test assembly including associated instrumentation and nondestructive examination plans
  - Examined effects of temperature on radiation induced volumetric expansion rates in concrete
  - Developed radiation-induced volumetric expansion/damage constitutive model
  - Evaluated advanced signal processing techniques to improve detection and identification for the nondestructive examination of concrete

Mitigation Technologies

- Transfer of weld-repair technique to industry (2018)
  - Completed weld model development and validation for laser welding process for the hot cell welding system
  - Completed friction stir welding process development for the hot cell welding system
- Complete development and testing of new advanced alloy with superior degradation resistance (2024)
  - Completed toughness testing and high temperature oxidation evaluations of advanced alloys for core internals
  - Completed screening of advanced alloys for core internals through ion-irradiation

Risk-Informed Safety Margin Characterization

The purpose of research and development work performed under the Risk-Informed Safety Margins Characterization (RISMC) Pathway is to support plant decisions for risk- informed margins management with the aim to improve economics and reliability, and to sustain safety of current nuclear power plants over periods of extended plant operations. The goals of the RISMC Pathway are twofold: (1) develop and demonstrate a risk-assessment method that is coupled to safety margin quantification that can be used by nuclear power plant decision makers as part of risk-informed margin management strategies; and (2) create an advanced RISMC Toolkit that enables more accurate representation of nuclear power plant safety margins and their associated impact on operations and economics.

Research Highlights

The research and development in this pathway falls into two categories: RISMC Toolkit development and RISMC Toolkit application. Select research and development highlights are provided here, followed by a list of major accomplishments (detailed reports covering the accomplishments can be found on the LWRS Program website: https://lwrs.inl.gov).

RISMC Toolkit Development

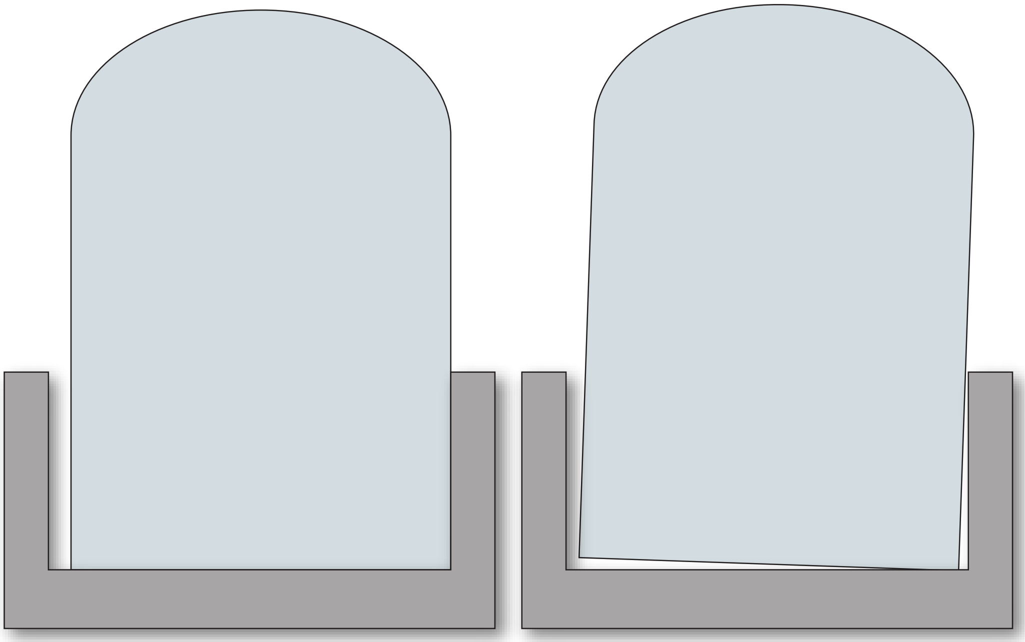
The RISMC Toolkit consists of a set of software tools that are used to perform the analysis steps in the RISMC method. The tools under development take advantage of advances in computational science and are based on (or are compatible with) a modern framework: the Multi-Physics Object Oriented Simulation Environment (MOOSE) developed at INL. These modern tools enable more efficient and more accurate modeling than is afforded by legacy tools.

RISMC Toolkit Development Highlight:

Mastodon Development. Mastodon is a MOOSE based application that models seismic soil-structure interaction (SSI) for nuclear power plants. It includes capabilities to simulate earthquake fault rupture, propagate the wave from the rupture site to the structure, and estimate the response of a structure or nuclear facility and the soil around it to the resulting seismic wave.

As the understanding of local seismology at a nuclear facility site evolves (for example: improved seismic source characterization, ground motion prediction equations, and local site effects), site-specific seismic hazard estimates may need to be updated. Larger ground motions result in increased soil strains; increased potential for gapping, sliding, and uplift between the structure and soil; and larger in-structure responses. Therefore, as the intensity of ground motions increases, the importance of appropriately capturing nonlinear effects in numerical SSI models increases.

In 2016, INL added capability to Mastodon to simulate 3-D wave passage effects through soil. Traditional calculations assume that soil behavior is "linear elastic." This means that numerically, as soil strains increase, there is no reduction in strength and corresponding increase in energy dissipation; therefore, as earthquake ground motion increases, structural response to that motion conservatively increases linearly. This can result in considerable conservatism in analyses. Mastodon includes nonlinear soil



Gapping and Sliding only *Gapping, sliding, and uplift*

Uplift is an important characteristic for seismic calculations and will be implemented in Mastodon in 2017.

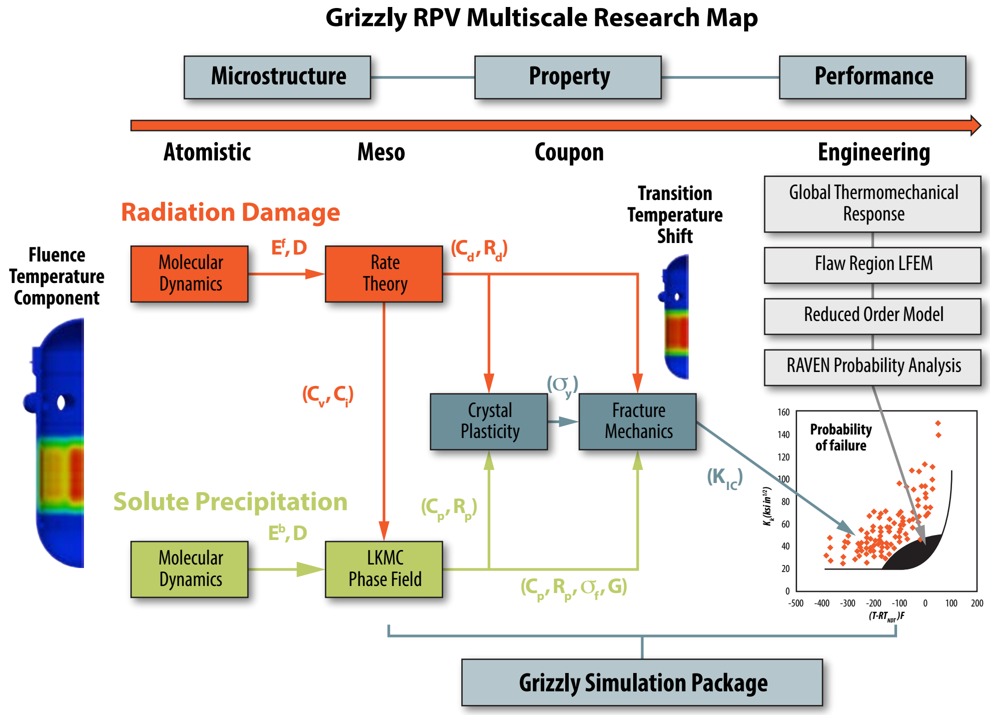
behavior resulting in a more accurate analysis of the wave propagation, important for overall accuracy in analyzing SSI. A collaborative verification effort with University of Illinois Civil Engineering department demonstrated the capability of Mastodon to model three dimensional wave passage in nonlinear soil. Additional capability will be added to Mastodon over the next year to implement a numerical model that calculates cyclic gapping, sliding, and uplift (illustrated in the figure above) for cyclic shaking, stochastic finite elements (directly calculate soil material property uncertainty in one model run, instead of using Monte Carlo), and frequency independent damping (used to dissipate energy when displacements are small). In addition, web-based verification will be implemented and user manuals developed. Verification of added capabilities will occur in parallel with code writing activities. A beta version of Mastodon will be released for testing purposes in 2017.

RISMC Toolkit Development Highlight:

Beta 1.0 Grizzly Release. INL, together with Oak Ridge National Laboratory and University of Tennessee, is developing the MOOSE-based Grizzly application to address aging issues in a variety of nuclear power plant systems, structures, and components. Grizzly is a multi-physics simulation tool for characterizing the behavior of nuclear power plant systems, structures and components subjected to a variety of age-related degradation mechanisms. Grizzly simulates both the progression of aging processes and the capacity of components to safely perform after being subjected to those aging processes.

Because of the central role of the RPV in a nuclear power plant and the difficulty involved in replacement or mitigation of operational effects, the RPV has been the initial target for capability development in Grizzly. However, Grizzly will ultimately

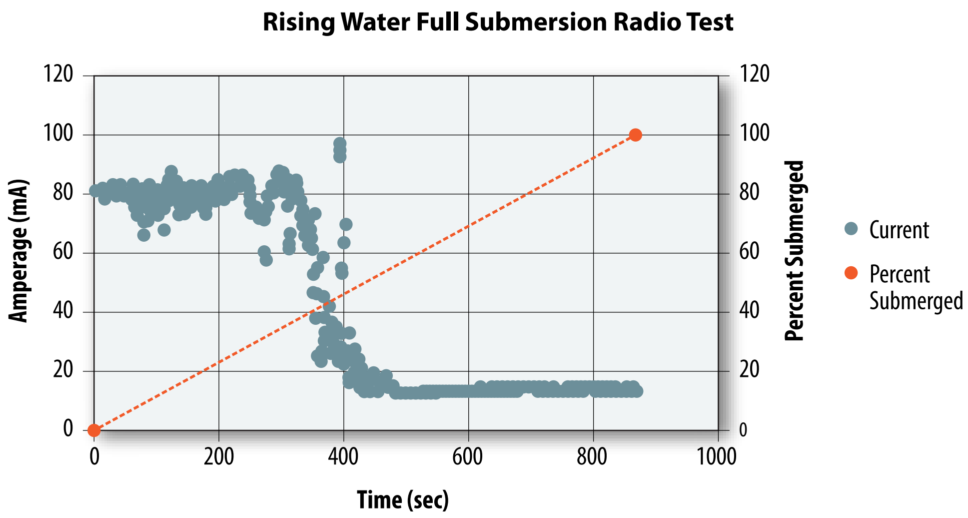
Grizzly is a multi-physics simulation tool for characterizing the behavior of nuclear power plant systems, structures and components subjected to a variety of age-related degradation mechanisms.



include capabilities for a wide range of components and materials. Work is also planned or ongoing to develop capabilities for modeling degradation in reinforced concrete structures, core internal structures, nickel-based alloys, and cables. The ability to simulate aging processes for steel, concrete, and cables will allow for investigation of risk-informed margins management over long periods of time when using the scenario-based tools being developed in the RISMC Pathway.

For the RPV application, Grizzly solves for the global thermal and mechanical response of an RPV under transient loading conditions and computes stress intensity factors in pre-existing flaws. Because RPVs are subjected to irradiation and elevated temperatures over time, the steel may become increasingly brittle, making it susceptible to fracture. Having the ability to predict the evolution of fracture toughness over time under those conditions is essential for predicting the strength and safety (for continued plant operation) of RPVs. Work is ongoing to develop models for evolution of the microstructure and fracture properties in Grizzly. The outcome of these lower-length-scale models will ultimately be used to predict the toughness of material exposed to fluences representative of operation to 60 years and beyond.

Because an RPV may contain a population of flaws with uncertain characteristics, probabilistic analyses must be used to determine its susceptibility to fracture over time. Grizzly is designed to interface with the RAVEN (Risk Analysis in a Virtual ENvironment) code to perform probabilistic fracture analysis of RPVs. The combination of these tools will provide the ability to perform probabilistic fracture analysis of RPVs with more general loading conditions and flaw geometry than the existing tools and will provide increased confidence in predictions for long-term operation.



Flooding fragility testing provides information on fragility that is used in flooding risk assessments.

The 1.0 Beta version of Grizzly and associated documentation was made available to a limited set of users and developers for the purposes of testing and obtaining feedback, including recommendations on additional feature development. This activity supports the completion of the RPV version of Grizzly in 2019 including the application of lower-length capabilities.

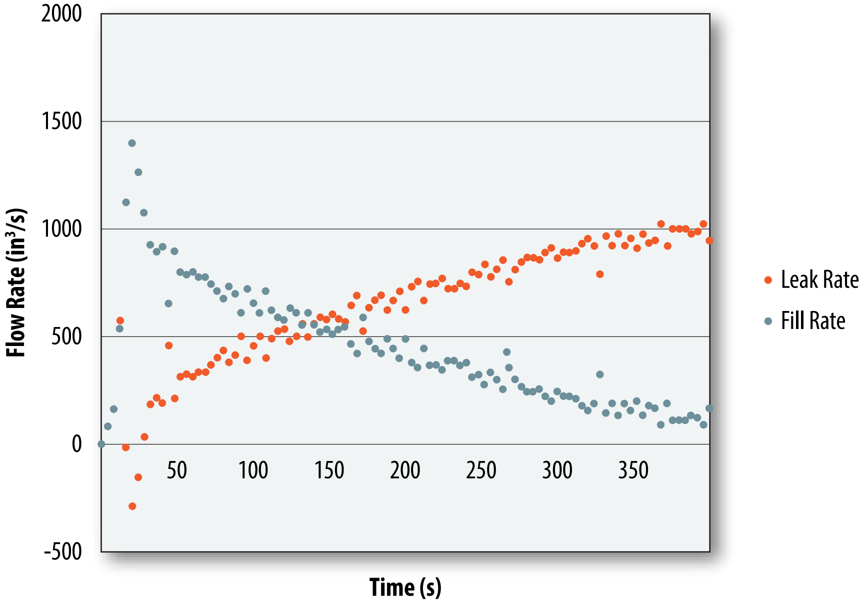
RISMC Toolkit Development Highlight:

Flooding Fragility Experiments. Flooding hazards are an important external event that plants must consider. Flooding hazards can be external (where water enters the plant site external to the plant, for example via a tsunami or river flood) or internal (where water inside the plant impacts structures or components in an off-normal condition, for example following a pipe break). One of the questions of interest in flooding-scenario modeling is how susceptible to failure structures and components are as a result of exposure to water ingress or spray. This susceptibility is characterized by the structure or component "flooding fragility." Historically, very simple conservative methods for conducting flooding risk assessment have been used that assume many components simply fail if contacted by water. Flooding fragility models are needed to accurately model the impact of flooding, potentially identifying additional safety margin.

To characterize flooding fragility, Idaho State University and INL tested a series of representative nuclear power plant components and structures to failure as part of a science-based approach to flooding risk analysis. Ultimately, wave impact, rising water, and top-down water spray testing will be done for both mechanical and electrical components. The experimental work is done in conjunction with flood simulation to conduct modeling-informed experimental design. The experimental data obtained will ultimately be used with the simulations for a more accurate flooding risk assessment. An example of the type of failure information that will be collected as part of the RISMC flooding experimentation is shown above.

Testing in 2016 included a series of door fragility tests with full-scale doors. In these experiments, doors were subjected to a series of water rise scenarios, and leakage,

Flooding fragility experiments are informing flooding risk assessments, potentially identifying additional safety margin relative to current methods.

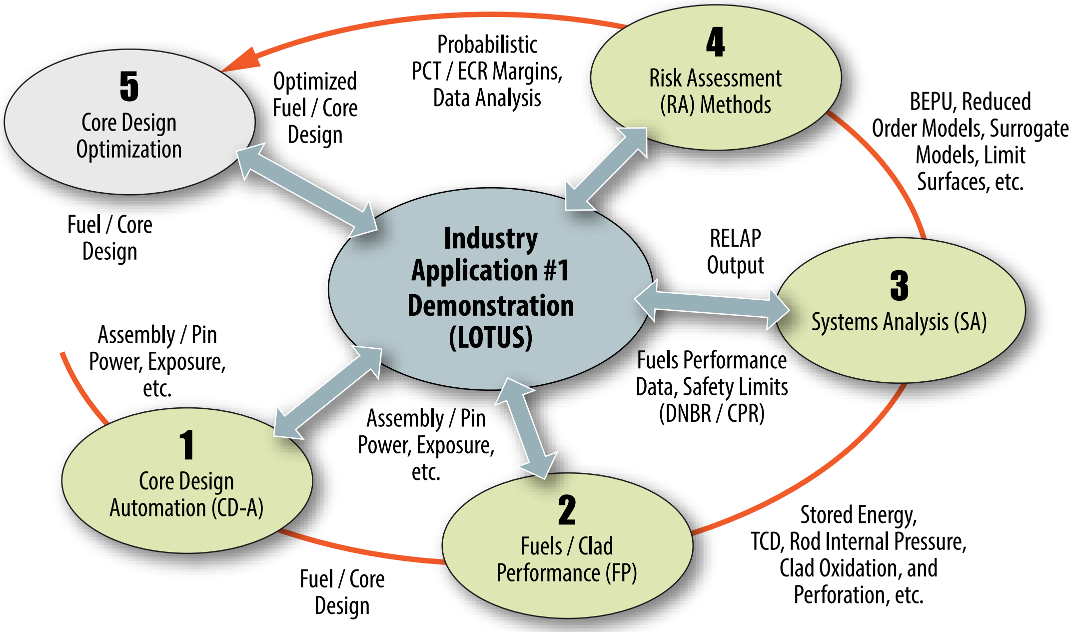


water depth in the tank, and inlet flow rate were measured. This set of experiments was also used to develop experimental protocol, informing future experiments. The component selected for the initial experiments was a hollow 3 × 7-foot tall wood interior door. A door frame and wall were constructed to hang the door in the tank. The door frame and wall was constructed using 2 × 6 inch lumbers and plywood. The studs were placed 9 inches apart and double studs were used next to the door. The standard code for building frame uses studs with 16-inch spacing. The studs were placed 9 inches apart to strengthen the wall and ensure the door was the weakest component subjected to the water rise. The figure above shows results of one of the series of tests.

This work supports the completion of flooding fragility experiments for mechanical components and a validated model for flooding analysis in 2018.

RISMC Toolkit Application

One of the primary avenues for collaboration with industry is through the RISMC Industry Applications. The primary purpose of Industry Applications in the RISMC



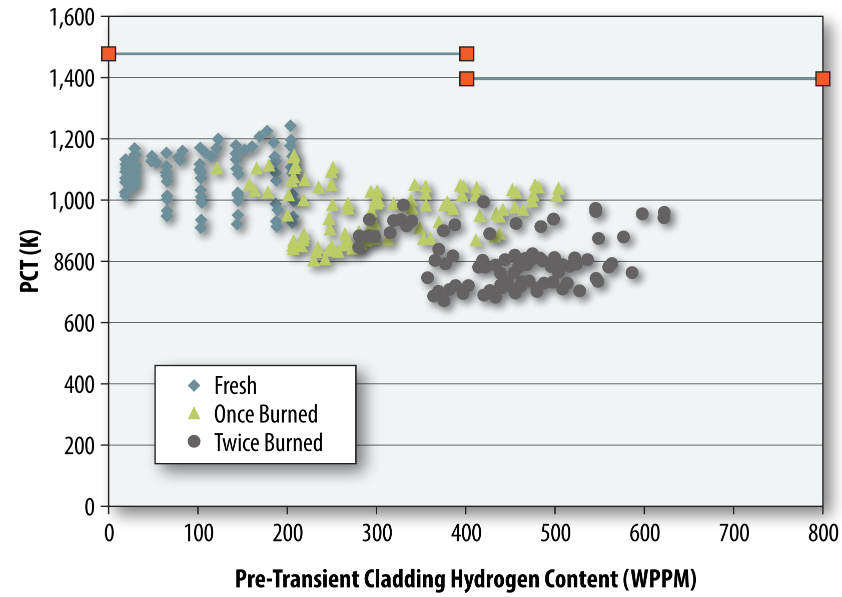
The LOCA Toolkit for the U.S. (LOTUS) will automate the integration of the multiple disciplines that are needed to evaluate a LOCA. BEPU is best-estimate plus uncertainty, and TCD is thermal conductivity degradation.

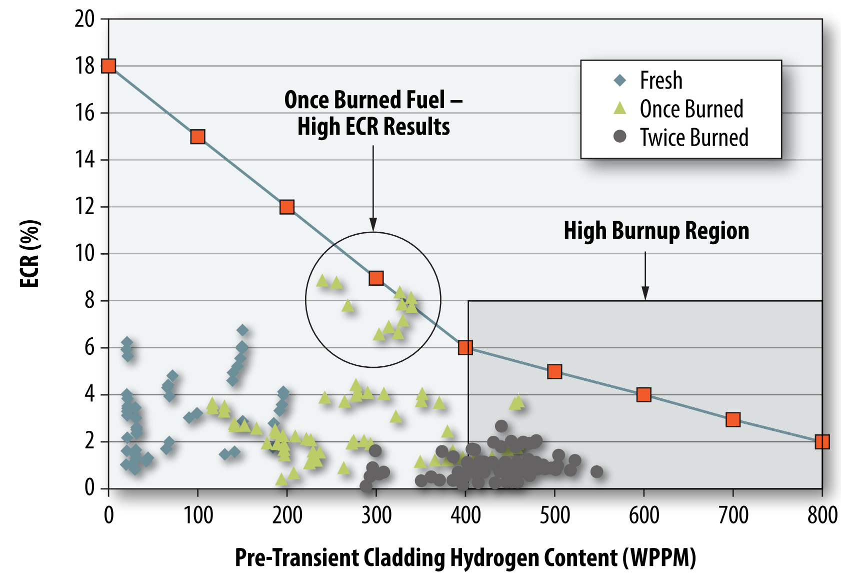
Pathway is to demonstrate advanced risk-informed decision making capabilities for relevant industry questions. The end goal of these activities is the full adoption of the RISMC tools and methods by industry applied to their decision making process. The RISMC Toolkit has progressed to the point where it can be applied to demonstration problems to illustrate the benefits from the RISMC methodology, using the modern computing tools under development in the LWRS Program and other Department of Energy programs. In addition to demonstrating the benefits of the RISMC methodology, these demonstration problems inform where additional model development is needed. Each of the industry applications will have one or more industry partners.

RISMC Toolkit Application Highlight:

Loss-of-Coolant Accident Analysis. The U.S. Nuclear Regulatory Commission (NRC) is proposing rulemaking 10 CFR 50.46c to revise the loss-of-coolant-accident (LOCA) and emergency core cooling system (ECCS) acceptance criteria to include the effects of higher burnup on cladding performance, resulting in a more restrictive rule than currently used. The key implications of this proposition are that the core, fuels, and cladding performance cannot be evaluated in isolation anymore. Cladding and ECCS performance need to be considered in a coupled manner. This may also suggest that models for cladding performance as well as LOCA methodologies need to be updated.

The RISMC methodology coupled with multi-physics modeling will provide plant owner/operators with a tool to inform decisions to manage margins related to compliance with the proposed 10 CFR 50.46c rule. In this project, margin is measured relative to the 10 CFR 50.46c proposed rule. Should the rulemaking proceed, the industry will need to comply with the proposed rule within 7 years of the proposed change (the timeline for implementation is still being discussed among the NRC, fuel vendors, and licensees, and will depend on many factors, such as methodology changes, amount of work to be submitted for regulatory approval, and regulatory reviews). A loss of operational margin may result due to the more restrictive cladding embrittlement criteria. Initial and future





The LOTUS approach provides important information relative to the proposed NRC rule 10 CFR 50.46c as well as for fuel cycle optimization. It introduces key fuels performance elements into the core design and reload analysis process.

compliance with the rule may significantly increase vendor workload and licensee costs, as a spectrum of fuel rod initial burnup states may need to be analyzed to demonstrate compliance. If the NRC decides not to go forward with rulemaking, this tool will still play an important role as it can be used for core optimization, resulting in cost savings. This multi-physics, multi-scale toolkit can also have an important role in analyzing high fuel burnup behavior, and other advanced fuel forms such as accident tolerant fuel. This methodology and tool (called LOTUS, for LOCA Toolkit for the U.S.) will quantify the impact on the key LOCA analysis figures-of-merit (peak cladding temperature (PCT), equivalent cladding reacted (ECR), and core-wide oxidation (CWO)) of a change in LOCA analysis inputs.

This approach will require less time than the approach used today. The information that the tool provides can then be used for decision-making and margin management. The benefits realized via this tool could outweigh the penalty associated with the proposed rule rollout. A more accurate analysis with respect to the actual margin available in each plant also can potentially reduce extensive (and costly) iterations between licensees and regulators when dealing with rule compliance and operational issues.

INL with support from Texas A&M University demonstrated the LOTUS methodology, including all aspects of LOTUS except for core optimization, which will be included in a future demonstration. This demonstration uses a generic reference PWR and a reference LOCA transient, and is carried out using legacy codes. The new, MOOSE-based applications will be used once they are sufficiently mature. In the figure on page 28, two regions of interest are identified for ECR results. On the right, a region with once burned fuel with ECR results approaching a proposed 50.46c limit (circled area); and on the right, a high burn-up region with pre-transient cladding hydrogen contents above 400 wppm (gray-shaded area). A closer look at the fuel performance history and behavior in the circled region might be warranted. This could indicate that core designs should be optimized to reduce the peaking for once burned fuel.

In 2017, this work will include an analysis of an operating PWR (in partnership with South Texas Plant and Texas A&M), and the use of advanced tools (for example, RELAP-7) to demonstrate the impact that these more accurate tools may have on the analysis of an existing operating nuclear power plant.

2016 Risk-Informed Safety Margin Characterization Accomplishments

A summary of the 2016 RISMC Pathway accomplishments is provided below. For each research area, the major 2016 accomplishments follow the primary out-year deliverable that they support.

- The RISMC margins analysis techniques and associated tools are an accepted approach for safety analysis support to plant decision-making (2020)
  - Completed RAVEN/Grizzly pressurized thermal shock analysis
  - Released Beta 1.0 RAVEN, including theory manual
  - Released Beta 1.0 Grizzly including an engineering fracture capability for reactor pressure vessels, an engineering model for embrittlement, and a modular architecture for modeling aging mechanisms
  - Completed plan to verify and validate multi-hazard risk-informed margin management methods and tools
  - Completed first set of flooding fragility experiments for mechanical components including fragility prediction and uncertainty
  - Released Beta 1.5 release RELAP-7 including improved closure relationships, steam/water properties, and LWR components
  - Integrated human reliability analysis models into the simulation-based framework for the RISMC Toolkit
  - Demonstrated RISMC approach, including the effects of higher burnup on cladding performance, as part of the LOCA/ECCS evaluation of risk-informed margins management strategies for a representative PWR
  - Completed a multi-hazard (seismic and flooding) analysis using the RISMC methodology

Advanced instrumentation, Information, and Control Systems Technologies

Efforts in the Advanced Instrumentation, Information, and Control (II&C) Systems Technologies Pathway address safe and efficient modernization of the current instrumentation and control technologies used in nuclear power plants. This is done through development and testing of new instrumentation and control technologies and advanced condition monitoring technologies for more automated and reliable plant operation. The research and development products are used to design and deploy new II&C technologies and systems in existing nuclear power plants that provide an enhanced understanding of plant operating conditions and available margins and improved response strategies and capabilities for operational events. The goals are to enhance nuclear safety, increase productivity, and improve overall plant performance. Pathway researchers work with nuclear utilities to develop instrumentation and control technologies and solutions to support the safe and reliable long-term operation of current nuclear power plants.

Research Highlights

Select research and development highlights are provided here, followed by a list of major accomplishments (detailed reports covering the accomplishments can be found on the LWRS Program website: https://lwrs.inl.gov).

Pilot Projects

The Advanced II&C Systems Technologies Pathway is conducting a series of capability building pilot projects to develop and test new technologies and capabilities that can be replicated and used by other nuclear power plants. Each pilot project has value individually, as well as collectively, by demonstrating the means to achieve long-term sustainability of II&C systems and technologies.

Pilot Project Highlight:

Control Room Modernization. In spite of a significant number of digital systems now having been implemented in operating nuclear power plants, there have been no large-scale changes to the layout or function of their control rooms. Nuclear utilities have understandably been reluctant to undertake significant control room upgrades or modernization projects in consideration of the costs, regulatory risks, and impacts on the large investment in procedures, training programs, and other support functions that may accompany large upgrades. Also, there is a general desire to retain the high degree of operator familiarity with the current control room arrangements, and thereby avoid potential human performance issues associated with control board configuration changes.

However, introducing digital systems into the control rooms creates opportunities for improvements in control room functions that are not possible with analog technology. These improvements are actually enabled by the new digital instrumentation and control systems even though many times these features go unused. This is especially true of what is called distributed control systems, in which plant parameters are digital variables that can be used and displayed in multiple ways that are beneficial to operators. This is in contrast to analog technology in which a parameter is generally available in just one place on the control boards.

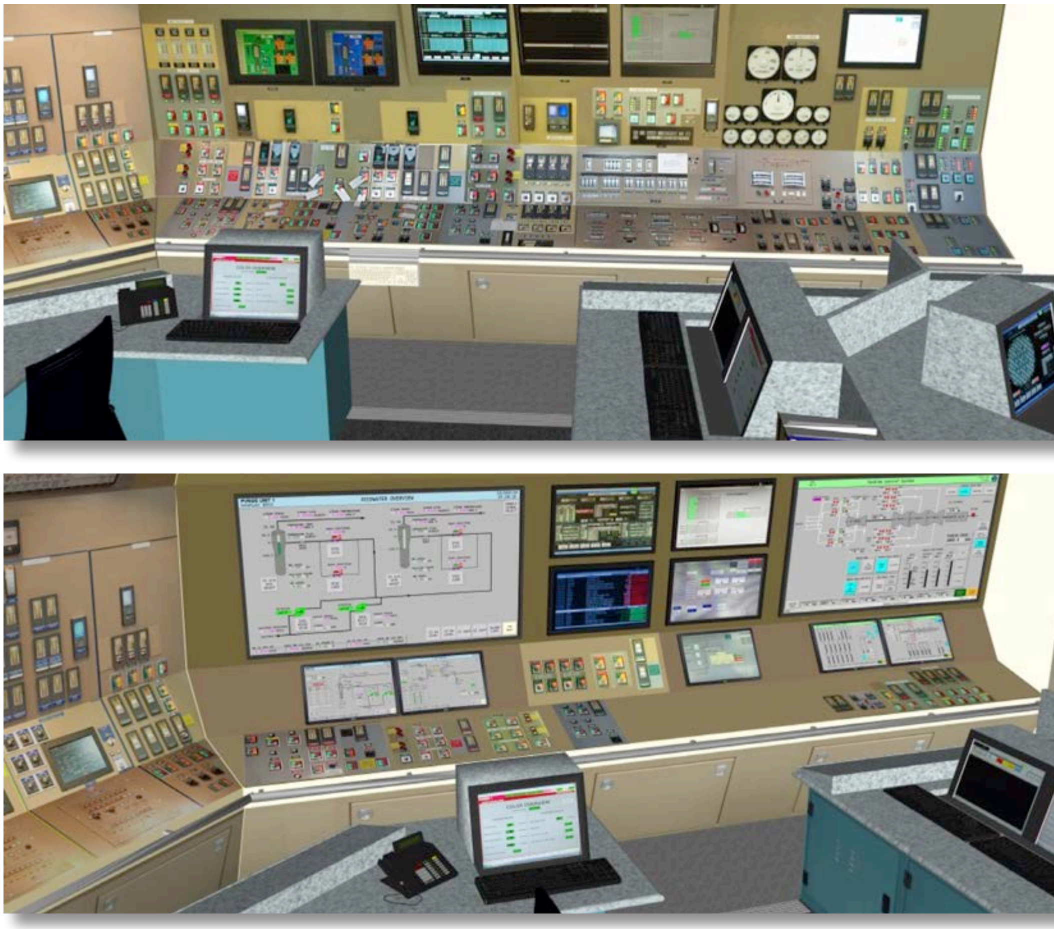
Palo Verde Nuclear Generating Station (Palo Verde) is undertaking a substantial instrumentation and control upgrade program, and is partnering with the LWRS Program to explore control room improvements that could be made in conjunction with planned system modernization efforts and how this can positively impact operator performance. The output from this multi-year study will enable other nuclear power plants to modernize their control rooms.

INL researchers, in collaboration with Palo Verde, developed an end-state vision for the Palo Verde main control room. The end-state vision provides the context by which to consider possible options to upgrade various control systems that will be replaced, so that the resulting integrated outcome is consistent with human factors engineering principles and provides substantial improvement in operator performance. Three methodologies with accompanying tools were used to develop and evaluate the end-state concept:

- Human Systems Simulation Laboratory (HSSL) – the use of the Palo Verde plant simulator and human-system interface prototypes to provide a functionally accurate tool for dynamic evaluation of the blend of new digital and legacy analog devices on the control boards.
- 3-D Modeling – development of dimensionally accurate models of the control room, both for the current state and various phases of the upgrade, to allow evaluation of human factors principles as well as certain functional and physical constraints.
- Virtual Reality: viewing of the 3-D models in the Computer-Assisted Virtual Environment to improve understanding of the spatial aspects of the control room.

Several human factors design and evaluation methods were used to develop the end-state concept, including interactive sessions with operators in the Human Systems Simulation Laboratory. Throughout these interactive sessions, human factors principles were considered that would affect the modification of the control boards, or how operator performance might be affected by certain design decisions. These effects might arise from, for example, the physical mounting of large overview displays, touch panels and other devices, or from the design of the distributed control systems displays. This review helped to confirm that all modifications must conform as far as practicable to human factors principles. This includes physical ergonomics (readability, viewing angles and reachability of new displays) as well as cognitive ergonomics (mental models, visual salience, visual complexity, information complexity and functional complexity). It also includes adhering to the basic principles of grouping, proximity, labeling, and association (i.e., keeping related components together).

One of the important changes envisioned for a more advanced control room is the change from an environment designed primarily for standing operations, to one that allows more tasks to be performed sitting down at the Reactor Operator and Senior Reactor Operator workstations. Ergonomic studies in other industrial environments have shown that standing-only work is unnecessarily strenuous for the duration of an eight-hour shift, and even worse for a twelve-hour shift, as is the



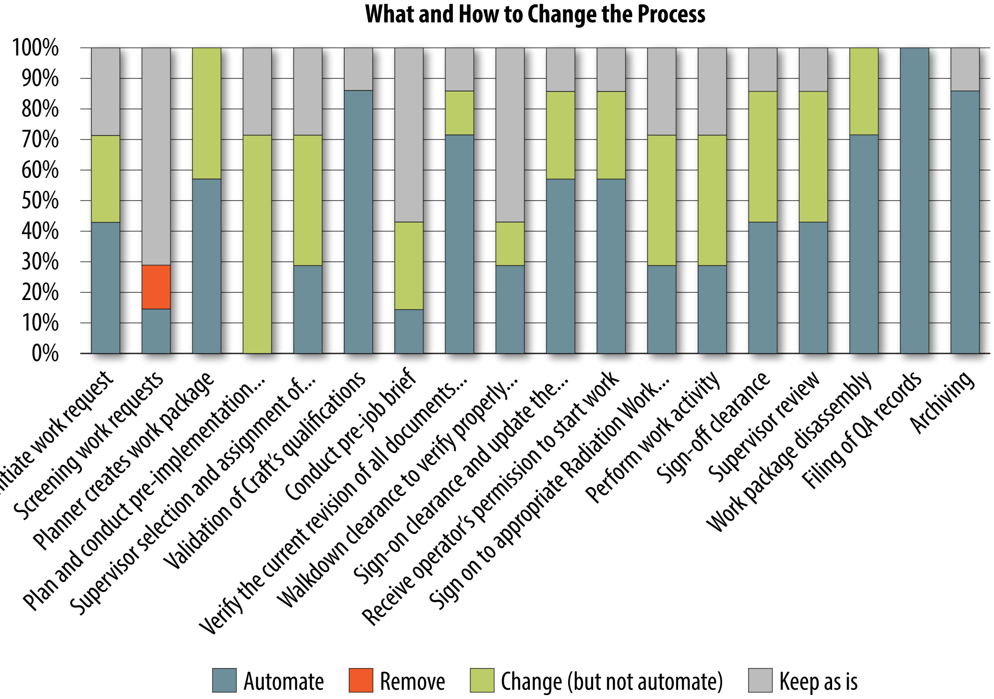
The graphic at top shows a typical analog control room. The second graphic shows how a control room can be transformed through digital upgrades.

practice in U.S. control rooms where operators perform the majority of their tasks at the control boards. The configuration under development will make it possible to monitor operations on the large overview display from that seated position. However, the ergonomic benefit must be confirmed through more detailed human factors assessments. A Human Factors Engineering Plan for the Palo Verde Control Room Modernization Project that describes a graded set of activities that will be used to ensure that the control room improvements conform to human factors principles will be completed in 2017.

This multi-year activity will be completed in 2025, with a published report on lessons learned and initial operational benefits. A companion activity to this project, fleet-based control room modernization, is in the planning stages with Exelon.

Pilot Project Highlight:

Automated Work Packages. Paper-based work packages for field activities are typically bulky and cumbersome. They are expensive and wasteful of paper to print, and the volume of paper can be overwhelming to transport to the job site and manage while there. Further, for activities in a radiation control zone, taking the needed and contingency paperwork to the job site often increases the amount of contaminated waste generated. Moreover, the paper-based work processes rely on human



Through a survey of personnel at nuclear utilities, targets of opportunity were identified for automated work packages.

performance to correctly obtain plant information, enter it into the work packages and procedures, successfully complete the steps of the process in the right sequence, and ultimately validate that the correct results have been obtained. Because of the complexity of these activities and the sheer bulk of the paperwork, errors can occur that cause incorrect final results, rework, time delays, excessive safety system unavailability, and latent nuclear safety issues, if errors go undiscovered.

An Automated Work Package (AWP) is an adaptive and interactive work package that intelligently drives the work process according to plant condition, resources status, and user progress. The AWP aims to enhance efficiency, enhance human performance, and reduce human errors by automating several manual tasks of the work process. Electronic work packages (eWPs) are work packages that rely, to various extents, on electronic data processing and presentation. AWPs are the logical evolution of eWPs. They are envisioned to incorporate advanced technologies and innovations of the future and address unresolved deficiencies in the work process of a nuclear power plant.

As an initial step in designing an AWP, INL researchers developed a scenario of the possible future work process without any current technology restrictions. The scenario targeted every stage of the work process execution. The scenario development resulted in 50 advanced functions that could be part of an AWP. To rank the importance of these functions, a survey was conducted that involved several U.S. nuclear utilities. The survey was aimed at determining the current needs of the nuclear industry with respect to the current work process (i.e., what the industry is satisfied with and where the industry envisions potential for improvement). The survey evaluated the most promising functions that resulted from the scenario development. The figure above shows the results of the survey. This information will provide input to priorities in the development of AWPs, including pilot projects at plant site. This work supports the completion of guidelines for implementing AWP capabilities for the nuclear power industry in 2019.

Pilot Project Highlight:

Computer Based Procedures. The paper-based procedures currently used for nearly all activities in the commercial nuclear power industry have a long history of ensuring safe operation of the plants. However, there is potential to greatly increase efficiency and safety (and therefore realize significant savings) by improving how the human interacts with the procedures. This can be achieved through the use of computer-based procedures (CBPs). A CBP system offers a variety of improvements, such as context-driven job aids, integrated human performance tools and dynamic step presentation.

The main purpose of the multi-year CBP research effort conducted by INL was to provide design guidance to the nuclear industry to be used by both utilities and vendors. Pilot projects were carried out at multiple plants: Duke Power's Catawba Nuclear Station, Pacific Gas & Electric's Diablo Canyon Power Plant, and Arizona Public Service's Palo Verde Nuclear Generating Station. After studying existing design guidance for CBP systems, the researchers concluded that the majority of the existing guidance is intended for control room CBP systems, and does not necessarily address the challenges of designing CBP systems for instructions carried out in the field. Further, the guidance is often presented on a high level, which leaves the designer to interpret what is meant by the guidance and how to specifically implement it. This design guidance is specifically tailored to instructions that are carried out in the field.

The high-level design requirements in the design guidance are:

1. Provide context sensitive information everywhere possible
2. Support all expected task flow characteristics

Computer-based procedures pilot projects were carried out at multiple nuclear power plants.

3. Support expected level of flexibility in performing task
4. Guide worker through logical sequence of the procedure
5. Provide information needed to control path through the procedure
6. Provide computerized support where appropriate and possible
7. Include functionality that improves communication
8. Provide a method to review and save records

The design guidance, the last activity in the LWRS Program specific to CBP, provides several specific examples of how to implement each of the high level requirements and provides illustrations and explanations of the observed benefits of the concepts.

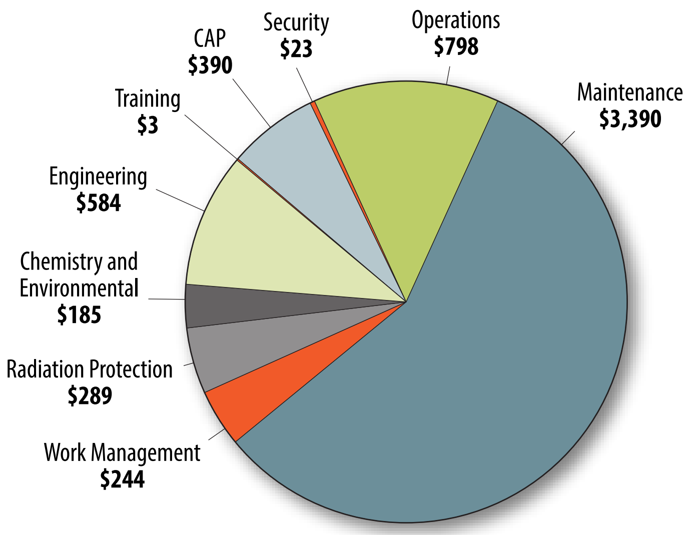
Pilot Project Highlight:

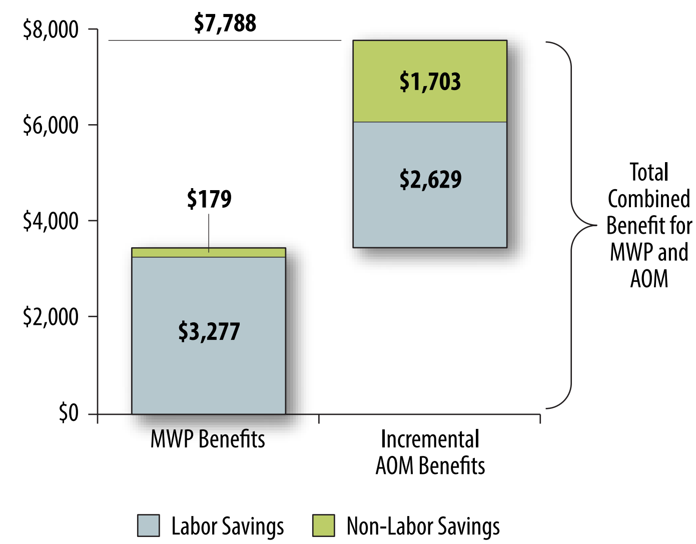
Business Case for Implementation of Technology Supporting Advanced Outage Management. The lack of a business case is often cited as a significant barrier in pursuing wide-scale application of digital technologies to nuclear power plant work activities. While the performance advantages of these new capabilities are widely acknowledged, it has been difficult for owner/operators to derive business cases that result in actual cost offsets that can be credited in budget allocations for site organizations, and reduce operations and maintenance costs. This is because the technologies are typically applied in a manner that simply enhances existing work methods rather than eliminating work or making it significantly more efficient. In 2015, INL, working with ScottMadden Management Consults (a firm that has years of experience in preparing performance improvement business cases for senior leadership in the nuclear power industry) developed a "Digital Technology Business Case Methodology." This methodology provides a structure for building the business case for adopting pilot project technologies in a manner that captures the total organizational benefits that can be derived from the improved work methods. This includes direct benefits in the targeted work processes, efficiencies gained in related work processes, and avoided costs through improvement in work quality and reduction in human error.

This year INL researchers (with ScottMadden Management Consultants) applied the methodology to implementation of technology that supports Advanced Outage Management (AOM). The analysis is built upon and incremental to a prior business case created for Mobile Work Packages (MWP). The business case demonstrates that advanced communication and networking and analytical technologies will allow nuclear power plants to conduct outages with fewer people in management roles, and with the remaining people more effective and more productive. The benefits are quantified to a rough order of magnitude that provides directional guidance to nuclear power plants that are interested in developing a similar business case.

AOM is enabled by a suite of technologies, including:

- High-bandwidth wireless networks
- Mobile devices
- Component identification technology
- Mobile wireless video cameras





The Business Case Methodology demonstrates substantial benefits in both labor and non-labor categories with the application of AOM.

- Touch enabled interactive displays
- CBP/AWP
- Intelligent plant configuration
- Advanced data analytics
- Micro-scheduling
- Meeting collaboration tools
- Team collaboration tools

An analysis was conducted to determine how these technologies might impact outage operations at a nuclear power plant. The analysis concluded that outage management capabilities would be improved with application of these technologies, and result in annual benefits for the plant, including:

- Automated status updates
- Advanced bulk work and schedule analysis
- Networked meetings (remote access)
- Networked emergent work teams
- Coordination of dispatchable resources
- Outage configuration management
- Remote job oversight
- Paperless outage coordination

The potential savings were quantified using the Business Case Methodology, showing annual savings of almost $8M/year as shown in the graphic below. The business case demonstrates substantial benefits in both labor and non-labor categories with the application of AOM.

2016 Advanced Instrumentation, Information, and Control Systems Technologies Accomplishments

A summary of the 2016 Advanced II&C Systems Technologies Pathway accomplishments is provided below. For each research area, the major 2016 accomplishments follow the primary out-year deliverable that they support.

- Health risk management framework for concrete structures in nuclear power plants (2018)
  - Developed a concrete health monitoring framework for alkali-silica reaction
  - Developed a unified approach to analyze and visualize heterogeneous data to support diagnosis and prognosis of alkali-silica reactions in concrete specimens
- Summary report on the Control Room Modernization Design Project providing lessons learned and initial operational benefits (2025)
  - Developed process to migrate from older to new digital system in a control room
  - Developed alarm displays and control room layout recommendations
  - Completed an operator study on the Palo Verde control room end-state
  - Developed methods to measure human performance in simulator studies using eye tracking
  - Developed a recommended end-state concept for the Palo Verde control room modernization design project
  - Complete installation and verify operability of Westinghouse computerized procedure systems (CPS) and alarm systems in the Human System Simulation Laboratory to support future research activities
  - Control room modernization business case that documents the quantitative and qualitative performance improvement potential
- Real-time outage risk management strategy to improve nuclear safety during outages by detecting configuration control problems caused by work activity interactions with changing system alignments (2019)
  - Completed business case for outage management technologies
  - Identified digital features required to integrate work order, procedures, mobile communication, and smart devices to achieve higher worker efficiency
  - Developed an overview display to allow advanced outage control center management to quickly evaluate outage status

Reactor Safety Technologies

The Reactor Safety Technologies Pathway is carrying out research and development efforts to improve understanding of beyond design basis events and reduce uncertainty in severe accident progression, phenomenology, and outcomes using existing analytical codes and information gleaned from severe accidents, in particular the March 2011 events at Fukushima Daiichi. This information will be used by industry to aid in developing mitigating strategies and improving severe accident management guidelines for the current light water reactor fleet.

Research Highlights

Select research and development highlights are provided here, followed by a list of major accomplishments (detailed reports covering the accomplishments can be found on the LWRS Program website: https://lwrs.inl.gov).

Fukushima Daiichi Forensics Examinations

Little is known about the end-state of core materials and key structures and components within Units 1, 2, and 3 at the Fukushima Daiichi Nuclear Power Station. However, similar to what occurred after the accident at Three Mile Island Unit 2, these Fukushima Daiichi units offer a unique means to obtain prototypic severe accident data from multiple full-scale boiling water reactor (BWR) cores related to fuel heat-up, cladding and other metallic structure oxidation and associated hydrogen production, fission product release and transport, and fuel/structure interactions from relocating fuel materials. In addition, these units may offer data related to the effects of salt water addition, vessel failure, ex-vessel core/concrete interactions, and Mark I drywell liner attack. Information obtained from these units not only offers the potential to reduce uncertainties in severe accident progression, but also may offer the potential for safety enhancements.

Information obtained from Fukushima Daiichi is required to inform decontamination and decommissioning activities, improving the ability of the Tokyo Electric Power Company (TEPCO) to characterize potential hazards and to ensure the safety of workers involved with cleanup activities. The Reactor Safety Technologies Pathway is using information obtained by TEPCO to enhance the safety of existing and future nuclear power plant designs. Significant safety insights are being obtained in the areas of component performance, fission product release and transport, debris end-state location, and combustible gas generation and transport. In addition to reducing uncertainties related to severe accident modeling progression, these insights are being used to update guidance for severe accident prevention, mitigation, and emergency planning. Furthermore, reduced uncertainties in modeling the events at Fukushima Daiichi will improve the realism of reactor safety evaluations and inform future decontamination and decommissioning activities by improving the capability for characterizing potential hazards to workers involved with cleanup activities.

Safety insights gained from the Reactor Safety Technologies Pathway forensics activities are being used by industry to update and improve BWR and pressurized water reactor (PWR) guidance for severe accident prevention, mitigation, and emergency planning in addition to reducing uncertainties in severe accident modeling progression. Examples below illustrate the impact of the forensics activities on severe accident guidance.

The units at Fukushima Daiichi that were in operation in March 2011 have, subsequent to the earthquake and resultant tsunami, exhibited different patterns of primary containment vessel leakage of fission products and hydrogen. The variability introduced by unit-to-unit differences points to uncertainties in actual leakage locations and confirms that maintaining containment conditions below design basis temperature and pressure limits (and that a high priority is placed on reducing containment conditions when they exceed design basis values) is an appropriate strategy. The BWR and PWR Owners Groups' revised severe accident guidance places a high priority on venting the primary containment when the combination of pressure and temperature reaches a prescribed limit. For BWRs, these conditions can be very close to the containment design basis pressure and temperature.

Currently available information indicates differences in the core debris end-state location in the three units. It is believed that these differences are due to differences in accident progression at each unit. The BWR and PWR Owners Groups' guidance places a higher priority on injection of water into the RPV compared to the primary containment. If the RPV fails, the injected water will flow through the RPV breach to the core debris in the primary containment. This ensures that core debris is cooled with injected water (and possibly submerged in water) regardless of its location. Because a large amount of water is required to cool the core debris in all possible locations (in the primary containment and in the RPV) for both BWRs and some PWRs, the emphasis on water addition in the updated guidance is appropriate. The BWR Owners Group also places a high priority on injection of water into the RPV using core spray to assist in more complete cooling of in-vessel core debris.

Reactor Core Isolation Cooling System Model

The Reactor Core Isolation Cooling (RCIC) System for BWRs and Auxiliary Feed Water (AFW) System for PWRs are the key safety systems that are used to remove decay heat from the reactor under a wide range of conditions ranging from operational pressures down to lower pressures approaching cold shutdown conditions. Both systems use steam produced from the reactor core decay heat to drive a steam turbine which in turn powers a pump to inject water back into the core (BWR) and into the steam generators (PWR) to maintain the needed water inventory for long-term core cooling.

Based on events at Fukushima Daiichi and subsequent analyses, it is known that RCIC operation was critical in delaying core damage for days (almost three days for Fukushima Daiichi Unit 2) even though the turbine-pump system ran without DC power for valve control and with high water temperatures from the BWR wetwell. The RCIC system apparently operated in a self-regulating mode supplying water to the core and maintaining core cooling until it eventually failed at about 72 hours. This indicates that there is significant margin in these emergency core cooling systems that has neither been quantified nor qualified with the NRC.

Except for loss-of-coolant accidents, where the primary system depressurizes down to containment pressure, RCIC and AFW are the major long-term heat removal systems employed under a wide range of transients and accidents for the two reactor types. All probabilistic risk assessment analyses indicate that the dominant accident sequences that are beyond design basis events (e.g., extended loss of AC power) would involve RCIC operation for BWRs and AFW operation for PWRs. Thus, extended performance of

RCIC and AFW systems under beyond design basis event conditions is very important to overall plant safety in terms of reducing both the likelihood and the consequences of core damage events involving extended loss of AC power.

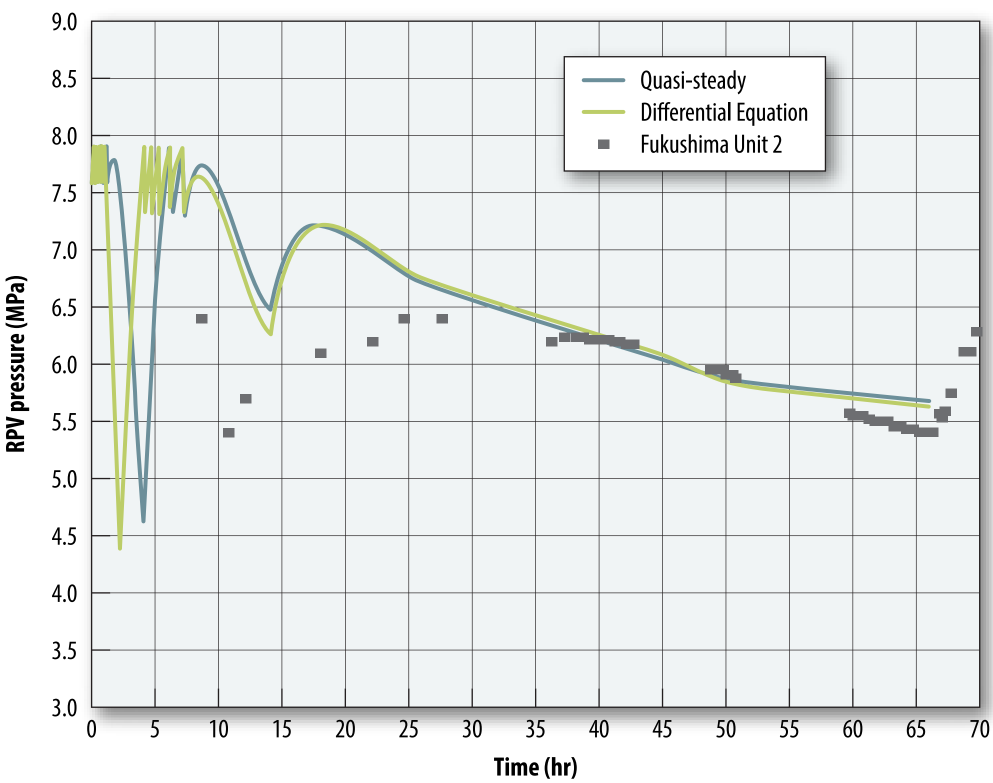
The first step towards quantifying/qualifying extended RCIC system performance with the NRC includes modeling RCIC system performance. Sandia National Laboratories is developing a dynamic and mechanistic system-level model of the Terry turbine-pump system that is capable of predicting system performance for RCIC and turbine-driven AFW applications under beyond-design-basis conditions. These applications include two-phase water ingestion into the Terry turbine at various potential reactor operating pressures and characterization of its ability to maintain adequate water injection with sufficient pump head under degraded operating conditions. This model will also demonstrate the self-regulating mode of operations as was observed in the Fukushima Daiichi Unit 2 accident, where RCIC ran uncontrolled and successfully maintained reactor water inventory for nearly 3 days.

Once developed, this "first principles" model can be refined through detailed computational fluid dynamics (CFD) models and can be incorporated into a systems-level code for simulation purposes.

Several analytical tools are being applied to investigate RCIC behavior for severe accidents. The tools include reactor system modeling codes, such as MELCOR and RELAP, and CFD codes, such as FLUENT. The primary goal is a mechanistic, system-level model that permits fast execution of long transient simulations (i.e., several hours to days for severe accidents). This will enable simulation capabilities for Fukushima Daiichi forensic analyses, development of technically defensible severe accident management guidelines/FLEX strategies, and design analysis of potential upcoming RCIC experiments. The reason to use several codes, both system-level and CFD, is to inform and enhance system-level modeling efforts using focused CFD analyses of key components, particularly where lumped-parameter methods and simple hand calculations have limited capability. An example is CFD analysis of the steam nozzles that drive the RCIC turbine. These detailed models more accurately represent nozzle performance.

A first principles model of the Terry turbine-pump system was developed for a RCIC BWR design and was tested by incorporating it into the MELCOR systems level code. To demonstrate its usefulness, an accident scenario that is comparable to Fukushima Daiichi Unit 2 was chosen. No 'tuning' or rigorous benchmarking against data was attempted here as there are still too many unknown and uncertain model parameters for such an effort to be meaningful. Moreover, available plant data are sparse. Instead, test calculations were deliberately performed for a non-Fukushima Daiichi model to demonstrate that the models have not just been forced to agree with the Fukushima Daiichi data. For example, the model demonstration uses an arbitrary power level of 2,000 MW and boiler properties from the Peach Bottom state-of-the-art reactor consequence analyses, including relatively high safety relief valve set points.

The figure on page 41 shows calculated RPV pressures compared to plant data for Fukushima Daiichi Unit 2. The green line represents the first principles model (differential equation model), which includes the complete differential equation with inertial terms for the Terry turbine-pump system; the blue line represents the



First-principles modeling of the RCIC system using MELCOR shows good agreement with trends indicated by data.

simplified first principles model (quasi-steady) incorporated into MELCOR, ignoring the inertial terms in the original differential equation model. The models are predicting key features of the RPV pressure trend that are in reasonable, qualitative agreement with plant data, despite the simple nature of the RCIC-MELCOR model and the deliberate modeling of a non-Fukushima Daiichi reactor.

Additional modeling is planned for 2017 as well as lab-scale experiments that will provide data for validating the models.

2016 Reactor Safety Technologies Accomplishments

A summary of the 2016 Reactor Safety Technologies Pathway accomplishments is provided below.

- Uncertainty analysis on Fukushima Daiichi events that provides particular areas of interest from a data sampling standpoint to be used in improving and validating severe accident codes
- Scope, cost estimates, and experimental plan development for expanding the operating band of the reactor core isolation cooling system
- Annual report on forensics data evaluations at Fukushima Daiichi
- Insights from severe accident analysis modeling for severe accident management guidelines

2017 DELIVERABLES PREVIEW

Building on the successes achieved in 2016, the LWRS Program has laid out an aggressive set of deliverables for 2017.

Materials Aging and Degradation

- Reactor Metals
  - Complete modeling of precipitate processes in irradiated reactor pressure vessel steel
  - Identify mechanisms for stress corrosion crack initiation in Alloy 600 and Alloy 690 exposed to PWR primary water environments
  - Complete fracture toughness characterization of reactor pressure alloys from the ATR-2 experiment
  - Testing to study the effect of swelling on irradiation assisted stress corrosion crack growth in high fluence steel
  - Complete microstructural analysis and mechanical testing of austenitic stainless steel samples irradiated at very high damage doses
  - Demonstrate computational tools to model combined thermal and radiation induced segregation of impurity solute elements to grain boundaries in austenitic stainless steels
  - Complete development and validation of LWR radiation-induced swelling model
  - Complete validated model for transition temperature shifts in RPV steels
  - Mechanical performance and microstructural evaluation of cast austenitic stainless steels following 10,000 hours aging
  - Results of stress corrosion crack initiation and propagation under different corrosion environments and the role of post-irradiation annealing on irradiation assisted stress corrosion crack mitigation
  - Deliver experimentally validated model of environmentally assisted fatigue in LWR components
  - Microstructural characterization of reactor pressure vessel alloys from the ATR-2 experiment
- Concrete
  - Submit concrete, minerals, and aggregates data to the Irradiated Minerals, Aggregates and Concrete (IMAC) Database
  - Monitoring data from embedded and external sensors of the alkali-silica reaction test assembly
  - Develop unified damage parameter for irradiated concrete constituents

Accelerated and naturally aged cable will be tested to evaluate impacts of degradation.

  - Quantify effects of various non-destructive examination parameters on the ability to detect and identify defects within structural concrete
  - Model the experimental alkali-silica reaction mockup data with Grizzly
- Cables
  - Complete tests with combined thermal/radiation aging at five dose rates in chlorosulfonated polyethylene (Hypalon)/ethylene-propylene rubber (EPR) cable jacket insulation system.
  - Analyze simultaneous thermal/gamma radiation aging of RSCC Firewall® III cross-linked polyethylene (XLPE) insulation
  - Analyze impact of accelerated thermal aging of Zion cable jacket and insulation between 80 and 120°C
  - Analyze baseline and initially aged Crystal River 3 cable materials subjected to simultaneous thermal/gamma radiation aging
- Mitigation Technologies
  - Complete cyclic plasticity material modeling of similar 316 stainless steel weldments and dissimilar 508 / 316 welds
  - High-temperature steam oxidation testing of select advanced replacement alloys for potential core internals
  - Fracture toughness evaluation of select advanced replacement alloys for core internals
  - Down-selection of candidate alloys for further testing in the LWRS and EPRI collaboration on advanced replacement materials
  - Develop welding parameters for irradiated materials

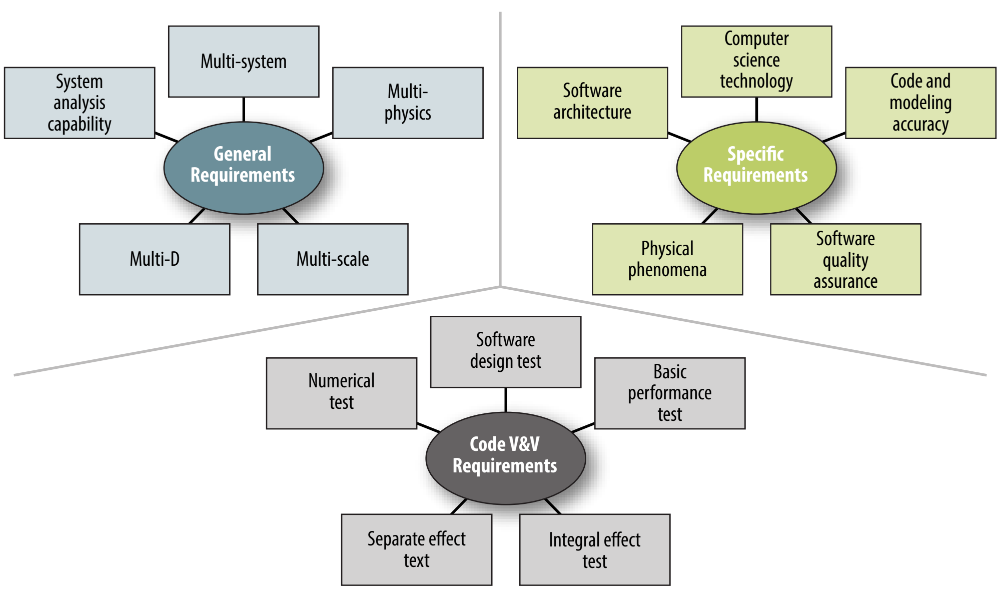
First monitoring data from embedded and external sensors of the alkali-silica reaction test assembly will be obtained.

Risk-Informed Safety Margin Characterization

- RISMC Toolkit
  - Develop and test 1-D downcomer and pressurizer components in RELAP-7
  - Demonstrate heat transfer phenomena of U-tube steam generator component and 1D pressurizer component for RELAP-7
  - Flooding fragility experiments with wall penetrations and door failures
  - Beta 1.0 release of seismic probabilistic risk assessment model
  - Beta 2.0 release including 3D sub-channel analysis capabilities
  - Beta 1.5 release of Grizzly (reactor metals)
  - Lower length scale modeling of embrittlement in reactor pressure vessel steels
- RISMC Applications
  - Model representative dynamic risk scenarios for a multi-unit power plant site using RAVEN
  - Integrated cladding/emergency core cooling system performance: Demonstration analysis of the South Texas Plant
  - Coupled margins analysis for enhanced external hazards including seismic and flooding phenomena
  - Full-scope margins analysis of a commercial reactor to analyze an industry-important issue such as application to 10 CFR 50.69
  - Apply RELAP-7 to loss-of-coolant accident thermal-hydraulic phenomena demonstrating vessel blowdown and loss of inventory

Advanced Instrumentation, Information, and Control Systems Technologies

- Plant model of the existing Exelon control room and a plant model of the Exelon planned upgrades in the Human Systems Simulation Laboratory
- Palo Verde control room modernization style guide, business case, human factors engineering plan
- Test vibro-acoustic and full field imaging techniques on medium sized concrete samples degraded with alkali-silica reaction

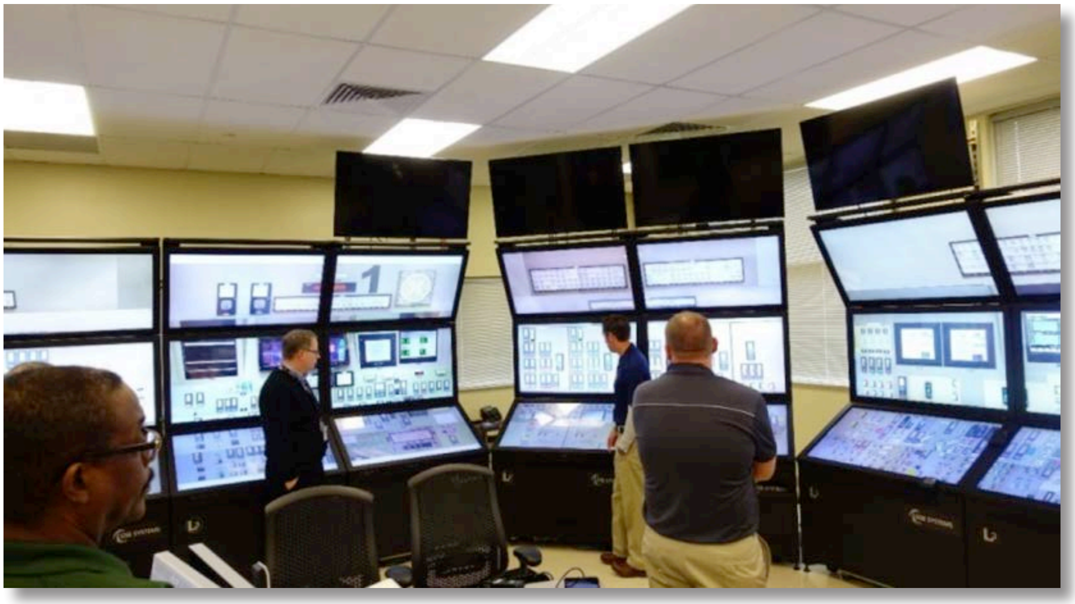


RELAP-7 verification and validation in 2017 will follow the RELAP-7 software verification and validation plan.

- Guidelines for regulatory aspects of the human factors engineering review processes for control room modernization
- Control room technology benefits study
- User study to evaluate the automated work package capabilities
- Evaluate the potential to extend the range of established online monitoring technologies, such as guided waves in nuclear power plant piping systems
- Information rich displays for a radiological waste control room at a nuclear power plant
- Development information rich displays for the main control room at a nuclear power plant collaborating with the fleet-based control room modernization design project.

Reactor Safety Technologies

- Compare RELAP-7 and MELCOR reactor core isolation cooling analyses
- Annual report documenting the results of forensics data evaluations at Fukushima Daiichi
- RELAP-7 Terry Turbine modeling enhancements
- Complete core debris cooling model (for industry use) including modeling, validation, and user's manual
- Complete MAAP/MELCOR crosswalk Phase 2 using an accident scenario that is similar to Three Mile Island-2 severe accident
- Ex-vessel severe accident analysis with the Melt Spread (MELTSPREAD) and Core Quench (CORQUENCH) codes in support of Severe Accident Guidance (SAG) development
- Complete melt spread model (for industry use) including modeling, validation, and user's manual
- Model and demonstrate the synergistic effects of accident tolerant fuels with extended reactor core isolation cooling operation on mitigation of beyond design basis accidents



Control room modernization activities use results of tests with actual plant operators in the Human Systems Simulation Laboratory.

PROGRAM CONTACTS

Program Management

Richard A. Reister

Federal Program Manager
Light Water Reactor Deployment
Office of Nuclear Energy
U.S. Department of Energy
richard.reister@nuclear.energy.gov
(301) 903-0234

Technical Integration

Kathryn A. McCarthy

Technical Integration Office Director
Idaho National Laboratory
kathryn.mccarthy@inl.gov
(208) 526-9392

Donald L. Williams

Technical Integration Office Deputy Director
Oak Ridge National Laboratory
williamsdljr@ornl.gov
(865) 574-8710

Cathy J. Barnard

Technical Integration Operations Officer
Idaho National Laboratory
cathy.barnard@inl.gov
(208) 526-0382

"The federal government's role is to support the sustainability of the nation's nuclear energy facilities by providing the science to enable the long-term safe, clean, and reliable operation of this important energy source through its unique facilities and expertise at DOE's national laboratories."

– **Richard Reister**
Federal Program Manager

Research and Development Pathway Leads

Materials Aging and Degradation

Keith J. Leonard
Oak Ridge National Laboratory
leonardk@ornl.gov
(865) 576-3687

Risk-informed Safety Margin Characterization

Curtis L. Smith
Idaho National Laboratory
curtis.smith@inl.gov
(208) 526-9804

Advanced Instrumentation, Information & Control Systems Technologies

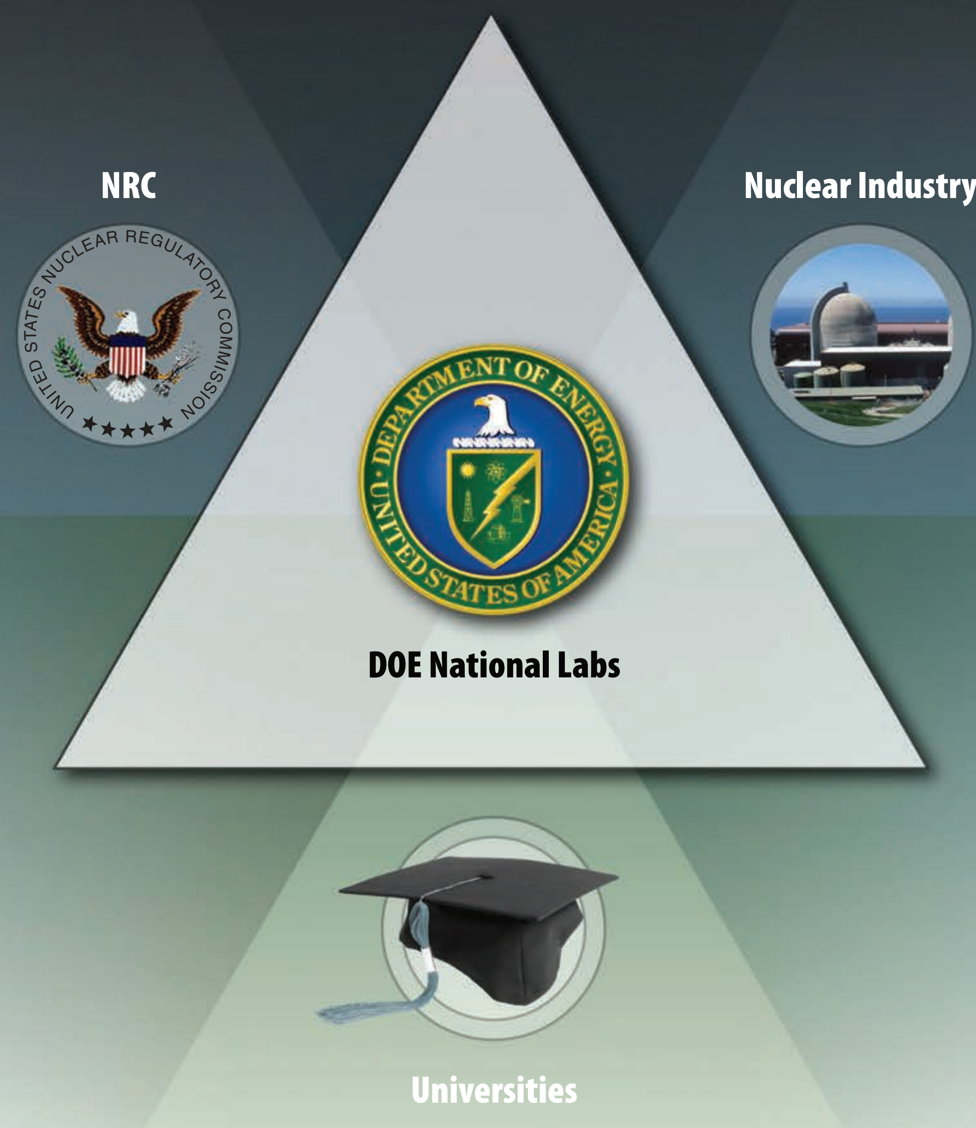
Bruce P. Hallbert
Idaho National Laboratory
bruce.hallbert@inl.gov
(208) 526-9867

Reactor Safety Technologies

Mitchell T. Farmer
Argonne National Laboratory
farmer@anl.gov
(630) 252-4539

LWRS
Light Water Reactor Sustainability

LIGHT WATER REACTOR SUSTAINABILITY PROGRAM



Working together to ensure energy security through the technically validated extended operation of nuclear power plants

16-GA50526